February 16, 1995

Mr. C. Randy Hutchinson
Vice President, Operations GGNS
Entergy Operations, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

SUBJECT: ISSUANCE OF AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M88078)

Dear Mr. Hutchinson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 119 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment revises the Technical Specifications (Tss) in response to your application dated October 22, 1993, as supplemented by letters dated February 10, and 14, 1995.

The amendment modifies the testing frequencies for the drywell bypass test and the airlock test, relocates certain drywell airlock tests from the technical specifications to administrative procedures, and incorporates various improvements from the Improved Standard Technical Specifications (NUREG-1434, Revision 0).

The staff has not completed its review of your request for performance based surveillance intervals up to 10 years for drywell bypass surveillance testing. Accordingly, your request will be deferred pending further discussion.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original signed by:
Paul W. O'Connor, Senior Project Manager Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures: 1. Amendment No. 119 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

February 16, 1995

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

MISSISSIPPI POWER AND LIGHT COMPANY

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated October 22, 1993, as supplemented by letters dated February 10, and 14, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-29 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 119 , are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W. O'Connor

Paul W. O'Connor, Senior Project Director
Project Directorate IV-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: February 16, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 119

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
1-2a	1-2a
3/4 6-13	3/4 6-13
3/4 6-14	3/4 6-14
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
B 3/4 6-3	B 3/4 6-3

DRYWELL INTEGRITY

1.10 DRYWELL INTEGRITY shall exist when:

a. All drywell penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE drywell automatic isolation system, or
   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

b. The drywell equipment hatch is closed and sealed.

c. The drywell airlock is in compliance with the requirements of Specification 3.6.2.3.

d. The drywell leakage rates are within the limits of Specification 3.6.2.1.

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each drywell penetration; e.g., welds, bellows or O-rings, is OPERABLE.

3/4.6.2 DRYWELL

DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.1 DRYWELL INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

Without DRYWELL INTEGRITY, restore DRYWELL INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 DRYWELL INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all drywell penetrations** not capable of being closed by OPERABLE drywell automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are opened under administrative control as permitted by Specification 3.6.4.

b. By verifying the drywell air lock is in compliance with the requirements of Specification 3.6.2.3.

c. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

d. By verifying drywell bypass leakage is less than or equal to 10% of the bypass leakage limit at least once per 18 months. (Not required to be performed until entry into MODE 2 on the first plant startup from the eighth refueling outage.)

** Except valves, blind flanges, and deactivated automatic valves which are located inside the drywell or containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

DRYWELL BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.2.2 Deleted

SURVEILLANCE REQUIREMENTS

4.6.2.2 Deleted

LIMITING CONDITION FOR OPERATION

3.6.2.3 The drywell airlock shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

NOTES

1. Entry and exit is permissible to perform repairs of the affected airlock components.

2. Required ACTION a is not applicable if both doors in the airlock are inoperable and ACTION c or d is entered.

3. Entry and exit is permissible for 7 days under administrative controls.

---

a. With one drywell airlock door inoperable, within 1 hour, verify the OPERABLE door is closed, and within 24 hours verified by administrative means to be locked closed at least once per 31 days. Otherwise, enter ACTION d.

NOTES

1. Required ACTION b is not applicable if both doors of the airlock are inoperable and ACTION c or d is entered.

2. Entry and exit is permissible under the control of a dedicated individual.

---

b. With the drywell airlock interlock mechanism inoperable, within 1 hour, verify an OPERABLE door is closed, and within 24 hours lock an OPERABLE door closed. Operation may continue provided that an OPERABLE door is verified by administrative means to be locked closed at least once per 31 days. Otherwise, enter ACTION d.

c. With the drywell airlock inoperable for reasons other than ACTIONs a or b, immediately initiate action to evaluate drywell overall leakage rate per LCO 3.6.2.1, "Drywell Integrity," using current airlock test results, and within 1 hour, verify a door is closed, and restore airlock to OPERABLE status within 24 hours. Otherwise, enter ACTION d.

d. With the required ACTIONs a, b, or c (including the associated completion times) not met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The drywell airlock shall be demonstrated OPERABLE:

a. Deleted

b. By conducting an overall airlock leakage test at ≥ 11.5 psig and verifying that the overall airlock leakage rate is ≤ 2 scfh at least once per 18 months.

NOTE

Only required to be performed upon entry into drywell.

---

c. At least once per 18 months by verifying that only one door in the airlock can be opened at a time.

d. Deleted

CONTAINMENT PURGE SYSTEM (Continued)

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failures develop. The 0.60 L_a leaking limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DRYWELL

3/4.6.2.1 DRYWELL INTEGRITY

Drywell integrity ensures that the steam released for the full spectrum of drywell pipe breaks is condensed inside the primary containment either by the suppression pool or by containment spray. By utilizing the suppression pool as a heat sink, energy released to the containment is minimized and the severity of the transient is reduced.

3/4.6.2.2 DRYWELL BYPASS LEAKAGE

The limitation on drywell bypass leakage rate ensures that the maximum leakage which could bypass the suppression pool during an accident would not result in the containment exceeding its design pressure of 15.0 psig. The design drywell leakage rate is expressed a $A/\sqrt{k}$ and has a value of 0.90 ft^2. $A/\sqrt{k}$ is dependent only on the geometry of drywell leakage paths where A = flow area of leakage paths in ft^2 and $\sqrt{k}$ is a lumped constant which considers geometric and friction loss coefficients such as discontinuities and Reynolds number. At a 3 psid differential pressure from drywell to containment an $A/\sqrt{k}$ of 0.90 ft^2 has an equivalent mass flow of 35,000 scfm. The integrated drywell leakage value is limited to 10% of the allowable drywell leakage capability, which is equivalent to 3500 scfm at 3 psid drywell to containment.

The $A/\sqrt{k}$ value of 0.90 ft^2 is derived from the analysis of "bypass capability with containment spray and heat sinks" (FSAR 6.2.1.1.5.5). The limiting case accident is a very small reactor coolant system break which will not automatically result in a reactor depressurization. The long term differential pressure created between the drywell and containment will result in a significant pressure buildup in the containment due to this bypass leakage.

3/4.6.2.3 DRYWELL AIR LOCK

The limitations on closure for the drywell air lock are required to meet the restrictions on DRYWELL INTEGRITY and the drywell leakage rate given in Specifications 3.6.2.1 and 3.6.2.2. The specification makes allowances for the fact that there may be long periods of time when the air lock will be in a closed and secured position during reactor operation. Only one closed door in the air lock is required to maintain the integrity of the drywell.

3/4.6.2.4 DRYWELL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the drywell will be maintained comparable to the original design specification for the life of the unit. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.2.5 DRYWELL INTERNAL PRESSURE

The limitations on drywell-to-containment differential pressure ensure that the drywell peak pressure of 22.0 psig does not exceed the design pressure of 30.0 psig and that the containment peak pressure of 11.5 psig does not exceed the design pressure of 15.0 psig during LOCA conditions. The maximum external drywell pressure differential is limited to +0.26 psid, well below the 2.3 psid at which suppression pool water will be forced over the weir wall and into the drywell. The limit of 2.0 psid for initial positive drywell to containment pressure will not allow clearing of the top vent which is consistent with the safety analysis.

3/4.6.2.6 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that peak drywell temperature does not exceed the design temperature of 330°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.2.7 DRYWELL VENT AND PURGE

The drywell vent and purge system must be normally maintained closed to eliminate a potential challenge to containment structural integrity due to a steam bypass of the suppression pool. Intermittent venting of the drywell is allowed for pressure control during OPERATIONAL CONDITIONS 1 and 2, but the cumulative time of venting is limited to 5 hours per year. Venting of the drywell is prohibited when either a 6-inch containment supply or exhaust valve or a 20-inch containment purge supply or exhaust valve is open, thus eliminating any resultant direct leakage path from the drywell to the environment.

Intermittent drywell venting and use of the drywell purge mode of the containment cooling system is allowed during OPERATIONAL CONDITION 3 to reduce the drywell airborne activity levels prior to and during personnel entry periods and to control drywell pressure, but is limited to 90 hours of use per 365 days.

3/4.6.3 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the drywell and containment pressure will not exceed the design pressure of 30 psig and 15 psig, respectively, during primary system blowdown from full operating pressure.

The suppression pool water volume must absorb the associated decay and structural sensible heat released during a reactor blowdown from 1060 psia. Using conservative parameter inputs, the maximum calculated containment pressure during and following a design basis accident is below the containment design pressure of 15 psig. Similarly the drywell pressure remains below the

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated October 22, 1993, as supplemented by letters dated February 10, and 14, 1995, the licensee (Entergy Operations, Inc.), submitted a request for changes to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TSs). The requested amendment would modify the testing frequencies for the drywell bypass test and the airlock test, relocate certain drywell airlock tests from the TSs to administrative procedures, and incorporate various improvements from the Improved Standard Technical Specifications (NUREG-1434, Revision 0).

The February 10, and 14, 1995, letters provided clarifying information that did not change the initial proposed No Significant Hazards Consideration determination.

Specifically, the licensee proposes the following changes to TS 1.10, 3/4.6.2.1, 3/4.6.2.2, and 3/4.6.2.3:

1. Modification of TS 1.10, definition for Drywell Integrity, to reflect the appropriate TS references.

2. Relocation of Limiting Condition of Operation (LCO) 3.6.2.2, Drywell Leakage Rate, into TS 3/4.6.2.1, Drywell Integrity, as a supporting surveillance.

3. Relocation of the details relating to drywell design from TS 3/4.6.2.2 to the TS Bases and the updated final safety analysis report (UFSAR).

4. Relocation of the details of the methods for carrying out the drywell bypass surveillance from TS 4.6.2.2 to the UFSAR.

5. Deletion of TS 4.6.2.2.c prohibiting use of the surveillance interval extension of 25% which would otherwise be allowed by TS 4.0.2.

6. Increasing the drywell bypass leakage test surveillance interval from 18 months to 10 years with an increased testing frequency required if performance degrades.

7. Modification of the TS LCO actions and surveillances for the drywell airlock in TS 3/4.6.2.3 to be consistent with NUREG-1434.

8. Relocation of TS 4.6.2.3.a, 4.6.2.3.d.2, and 4.6.3.d.3, pertaining to drywell airlock seal leakage and operability, to the UFSAR.

2.0 BACKGROUND

For several years, the NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TSs. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3288). During 1989 through 1992, the utility Owners Groups and the NRC staff developed improved Standard Technical Specifications (STSs) that would establish models of the Commission's policy for each primary reactor type. In addition, the staff, licensees, and the Owners Groups developed generic administrative and editorial guidelines in the form of a "Writers Guide" for TSs, which affords a significant enhancement of human factors considerations and was used throughout the development of licensee-specific improved TSs.

In September 1992, the Commission issued NUREG-1434, which was developed utilizing the guidance and criteria contained in the Commission's interim policy statement. It was established as a model for developing improved TSs for the BWR/6 plants in general and for the improved Grand Gulf Nuclear Station TSs specifically. NUREG-1434 reflects the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which were published in a "Split Report" issued to the NSSS Owners Groups in May 1988. NUREG-1434 also reflects the results of extensive discussions on various drafts of STSs, so that the application of the TS criteria and the Writers Guide would consistently reflect detailed system configurations and operating characteristics for all NSSS designs. As such, the generic Bases presented in NUREG-1434 provide an abundance of information regarding the extent to which the standard technical specifications present requirements which are necessary to protect the public health and safety.

On July 22, 1993, the Commission issued its Final Policy Statement. Therein, the Commission expressed its view that satisfying the guidance in the policy statement also satisfies section 182a of the Atomic Energy Act and 10 CFR 50.36. The Final Policy Statement described the safety benefits of the improved STSs and encouraged licensees to use the improved STSs as the basis for plant specific TS amendments, and for complete conversions to improved STSs.

Further, the Final Policy Statement provided guidance to evaluate the required scope of the TSs, and finalized the guidance criteria to be used in determining which of the design conditions and associated surveillances need to be located in the TSs. The Commission noted (58 FR at 39136) that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TSs, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

In accordance with this approach, existing TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement should be retained in the TSs, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents. The Final Policy Statement criteria are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.[1]

In its license amendment application, the licensee proposed changes to existing TS requirements using the Final Policy Statement and NUREG-1434 as guidance.

In this safety evaluation (SE), the licensee's proposed changes to its existing TS requirements are grouped into four general categories as follows: administrative, i.e., non-technical changes; relocated requirements, i.e., movement of requirements from existing TSs (an NRC-controlled document) to specified licensee-controlled documents; more restrictive requirements, i.e., additions to existing TS; and less restrictive requirements, i.e., relaxations to, or deletions from existing TS requirements. These four general categories of changes to the licensee's existing TS requirements may be better understood as follows.

Administrative Changes

Non-technical, administrative changes were intended to incorporate human-factors principles into the form and structure of the improved plant TSs so that they would be easier to use for plant operations personnel. These changes are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content or operational requirements. In order to ensure consistency, the NRC staff and the various licensees of the BWR/6 conversion plants have used NUREG-1434 as guidance to reformat and make other administrative changes. The licensees proposed such changes as: (a) providing the appropriate numbers, etc., for NUREG-1434 bracketed information (information which must be supplied on a plant-specific basis, and which may change from plant to plant), (b) identifying plant-specific wording for system names, etc., and (c) changing NUREG-1434 section wording to conform to existing licensee practices.

The staff has reviewed all of the administrative and editorial changes proposed by the licensee and finds them acceptable, since they are compatible with the "Writers Guide" and NUREG-1434, and are consistent with the Commission's regulations.

---

[1] The Commission recently promulgated a proposed change to 10 CFR 50.36, pursuant to which the rule would be amended to codify and incorporate these criteria (59 FR 48180). The Commission's Final Policy Statement specified that Reactor Core Isolation Cooling, Isolation Condenser, Residual Heat Removal, Standby Liquid Control, and Recirculation Pump Trip are included in the TS under Criterion 4. In the proposed change to §50.36, the Commission specifically requested public comments regarding application of Criterion 4. Until additional guidance has been developed, Criterion 4 will not be applied to add TS restrictions other than those indicated above.

Relocated Requirements

As summarized above, the Commission's policy statement provides that existing TS requirements which do not satisfy or fall within any of the four specified criteria may be relocated to appropriate licensee-controlled documents. In the licensee's application, such requirements are generally relocated to the UFSAR. Provisions of the existing TS action statements and surveillance requirements (SRs) will be relocated to appropriate plant procedures; i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures, depending on the nature of the requirements being relocated. Any time the operability of a system or component has been affected by repair, maintenance or replacement of a component, plant procedures require that a post-maintenance test be performed to demonstrate operability of the system or component. The existing TSs have various post-maintenance surveillance requirements distributed throughout which have been relocated from the improved TSs. In addition, the details and methods of operation of a system during the performance of a surveillance have been relocated from the existing TSs. Examples include descriptions of tests to assure controls of the system are operable, controls during functional testing of components, and setpoint verification which inherently performs a functional test of the instruments and the cycling of valves. These procedures will similarly be described in the UFSAR.

The facility and procedures described in the UFSAR can only be revised in accordance with the provisions of 10 CFR 50.59, which ensures an auditable and appropriate control over the relocated requirements and any future changes to these provisions.

Although the UFSAR already includes most of the design information described above, by letter dated November 10, 1994, the licensee committed to confirm that these details are appropriately reflected in the UFSAR, improved TS Bases or will be included in the next update of these documents. The licensee has also committed to maintain an auditable record of and an implementation schedule for the procedure changes associated with the development of the improved plant-specific TSs. The documentation of these changes will be maintained by the licensee in accordance with the record retention requirements specified in the QA Plan.

As described in more detail in this evaluation, the staff concludes that appropriate controls have been identified for all of the requirements that are being relocated from the licensee's TSs to licensee-controlled documents. Until incorporated in the UFSAR and procedures, changes to the provisions being relocated from the TSs will be controlled in accordance with the applicable existing procedures that control these documents. The staff concludes that, in accordance with the Commission's policy statement, sufficient regulatory controls exist under the regulations, particularly 10 CFR 50.59, to assure continued protection of the public health and safety. Accordingly, the staff concludes that these requirements, as described in detail in this evaluation, may be relocated from the TSs to the UFSAR or to other licensee-controlled documents as specified herein.

Less Restrictive Requirements

Less restrictive requirements are justified on a case-by-case basis as discussed in Section III of this evaluation. When requirements have been shown to provide little or no safety benefit, their removal from the TSs may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (a) generic NRC actions, (b) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups' comments on the improved STSs. Generic relaxations contained in NUREG-1434 were reviewed by the staff and found to be acceptable because they are consistent with current licensing practices and NRC regulations. The licensee's design was reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in NUREG-1434 and thus provide a basis for these revised TSs.

The following sections explain the staff's reasons for concluding that the conversion of the licensee's existing TSs to improved TSs based on NUREG-1434, as modified by plant specific changes, is consistent with the current plant specific licensing basis, applicable regulatory requirements and guidance of the policy statement, and is acceptable.

3.0 EVALUATION

A. Significant Administrative Changes

In accordance with the guidance in the Final Policy Statement, the licensee has proposed administrative changes to the existing TSs to bring them into conformance with the improved TS. These changes are as follows:

Change No. 1. This change is administrative in nature. It changes the reference in TS Section i, Definitions, to reflect the relocation of the drywell leakage rate requirements from TS 3.6.2.2 to TS 3/4.6.2.1. This change is administrative and maintains the TS Definition references consistent with the relocated TS designations. This change is acceptable to the staff.

Change No. 3. This change deletes the value of the acceptable $A/\sqrt{k}$ from TS 3/4.6.2.2. The allowable drywell bypass leakage limit of 0.90 ft^2 is derived from the analysis of "bypass capability with containment spray and heat sinks" contained in UFSAR Section 6.2.1.1.5.5 and is included in TS Bases Section 3/4.6.2.2 and the UFSAR. This change is administrative and is consistent with the format established in NUREG-1434 and is acceptable to the staff.

Change No 4. This change relocates the detailed methods for carrying out the drywell bypass surveillance from TS 4.6.2.2 to the UFSAR. As stated above, the requirement that the drywell leakage not exceed 10% of the allowable drywell bypass leakage limit is being relocated without change to TS 4.6.2.1. This change, which relocates surveillance details, is administrative in nature and consistent with the format of NUREG-1434 and is acceptable to the staff.

Change No. 7. This change modifies the TS LCO actions and surveillances for the drywell airlock in TS 3/4.6.2.3 to be consistent with the format and content of NUREG-1434.

7b. The revised presentation of actions in the proposed TSs do not propose to explicitly detail the most obvious option "to restore .... to operable status." This action, stated in the existing TSs, is not repeated in the improved TSs because this option is implicit in all conditions. Therefore, this provision is unnecessary and omitting this action is purely editorial.

7g. The reference to Special Test Exception 3.10.1 in TS 3.6.2.3 is proposed to be deleted in accordance with the guidance in NUREG-1343. No "cross references" are provided in the improved format TSs. These types of references serve no functional purposes and therefore, removal is purely an administrative preference in presentation which has no safety significance.

7j. SR 4.6.2.3.b.2 is proposed to be deleted. SR 4.0.1 requires that SRs must be continued to be met in the operational modes in which operability of a component is required. Therefore, if the operability of a component has been affected by repair or maintenance, post maintenance testing is required to demonstrate operability of the component or system. The staff finds that this deletion is administrative only because post maintenance testing is required elsewhere in the TSs.

The above changes are considered purely administrative and are therefore acceptable.

B. Relocated Requirements

Change No. 2. This change relocates the requirement that the drywell bypass leakage not exceed 10% of the drywell bypass leakage limit from LCO 3.6.2.2,"Drywell Bypass Leakage" to SR 4.6.2.1 that demonstrates that drywell integrity is being maintained. The allowed drywell bypass leakage remains unchanged at 10% of the bypass leakage limit. This change is consistent with the format established in NUREG-1434 for General Electric BWR/6 plants and is therefore acceptable to the staff.

Change No. 7. This change modifies the TS LCO actions and surveillances for the drywell airlock in TS 3/4.6.2.3 to be consistent with the format and content of NUREG-1434.

7.a The details comprising operability and testing of the drywell airlock currently in TS 3/4 3.6.2.3 are proposed to be relocated into the TS Bases and the UFSAR. The operability requirements for the drywell airlock door interlocks and details relating to the methods of surveillance are explicitly required in the current airlock operability and surveillance TSs. The details of the

methods and acceptance values for these continuity and system functional tests are located in and adequately controlled by plant procedures and improved TS Bases. The values are the original system design values and are also controlled by the design change procedures and 10 CFR 50.59.

7d. Current ACTION c. and surveillance 4.6.2.3.d.1 are proposed to be relocated. The airlock inflatable seal pressure instrumentation channels and associated alarm do not necessarily relate directly to airlock operability. The BWR/6 STSs in NUREG-1434 do not specify indication-only or alarm-only equipment required to be operable to support operability of a system or component. Control of the availability of, and the necessary compensatory activities required in the event the equipment is not available is contained in the associated plant procedures and operating policies or in the TS Bases.

Change No. 8. The surveillances pertaining to airlock seal leakage and operability in existing TS 4.6.2.3.a, 4.6.2.3.d.2, and 4.6.2.3.d.3 are proposed to be relocated to the UFSAR.

TS 4.6.2.3.a requires that the drywell airlock be demonstrated operable within 72 hours after each closing, except when the airlock is being used for multiple entries, then at least once per 72 hours, by verifying the seal leakage rate to be less than or equal to 2 scf per hour when the gap between the door seals is pressurized to P_a, 11.5 psig.

TS 4.6.2.3.d.2 requires that the drywell airlock be demonstrated operable at least once per 7 days by verifying each airlock door inflatable seal system operable by verifying seal air flask pressure to be greater than or equal to 90 psig.

TS 4.6.2.3.d.3 requires that the drywell airlock be demonstrated operable by verifying each airlock door inflatable seal system operable by, at least once per 18 months, conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 2 psig from 90 psig within 48 hours.

These three surveillances related to the airlock seal leakage and operability are redundant to the surveillance required by TS 4.6.2.1.d which requires that the total drywell leakage rate be measured periodically to assure that the design basis drywell allowable leakage rate of 35,000 scfm is not exceeded.

The licensee has performed an engineering review to establish the worst case leakage through the drywell airlock in the worst case assumption that one airlock door is open and both seals on the closed airlock door are failed. This worst case review, assuming a differential pressure of 3 psi, determined that the maximum leakage rate would be 5,000 scfm. The maximum allowed drywell leakage rate is set at 3500 scfm (10% of the

design basis drywell allowable leakage rate). The maximum measured drywell leakage rate since initial operation was 2599 scfm and the average leak rate of the 10 tests since initial operation is 1492 scfm. If the 5000 scfm worst case airlock leakage rate is added to the maximum allowable drywell bypass leakage rate of 3500 scfm, the total drywell leakage rate is less than or equal to 8500 scfm, which is less than 25% of the design basis allowable drywell bypass leakage rate. This calculation is conservative because the maximum allowable drywell leakage rate already accounts for actual leakage through the airlock seals because the surveillance is conducted with one drywell airlock door open. Based on the above evaluation the staff finds the relocation of TS 4.6.2.3.a, 4.6.2.3.d.2, and 4.6.2.3.d.3 acceptable.

The above relocated requirements relating to containment systems are not required to be in the TSs under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed in the Introduction above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59 or TS 5.5.11 to assure continued protection of the public health and safety. Accordingly, the staff has concluded that these requirements may be relocated from the TSs to the licensee's TS Bases, UFSAR or plant procedures, as applicable.

C. Less Restrictive Requirements

Change No. 5. This change deletes TS 4.6.2.2.c prohibiting use of the surveillance interval extension of 25% which would otherwise be allowed by TS 4.0.2. The change will permit the surveillance to be carried out on an interval up to 25% longer than specified in TS 4.6.2.1. This change, which relocates surveillance details, is consistent with the provisions of NUREG-1434 with regard to Drywell Bypass Leakage Surveillance, will avoid an unnecessary reactor shutdown to carry out this surveillance, and is acceptable to the staff.

Change No. 6. This change increases the drywell bypass leakage test surveillance interval from 18 months to 10 years with an increased testing frequency required if performance degrades.

The Mark III containment design at the GGNS, incorporates the drywell/pressure-suppression features of previous BWR containment designs into a dry containment structure. The function of the drywell is to force steam generated from a LOCA through the weir wall vents into the suppression pool, so it can be condensed. Any steam that bypasses the suppression pool and directly enters the dry containment structure has the potential to rapidly increase the containment pressure. The pressure-suppression capability of the suppression pool assures that the peak LOCA temperature and pressure in the primary containment are kept below the design limits of 185°F and 15 psig. Since the structural integrity of the primary containment is largely dependent

on the drywell's ability to perform its safety function, the total drywell bypass leakage area must be monitored.

GGNS TS 4.6.2.2, "Drywell Bypass Leakage," requires that a drywell bypass leakage rate test (DBLRT) be performed at least once every 18 months to verify that the steam bypass leakage area is less than or equal to 10% of the maximum allowable leakage path area of 0.90 ft^2.

By letter dated October 22, 1993, the licensee proposed revising the test frequency of the DBLRT based on a performance-based approach. DBLRT frequency would be extended up to once every 10 years. The frequency would be increased to once every 36 months following a test failure but could be reestablished at 10 years, if the next test was successful. If two consecutive DBLRTs failed to meet the acceptance criteria, a DBLRT must be performed at least once every 18 months, until two consecutive tests meet the acceptance criteria. The staff has not completed its review of this proposal to increase the surveillance interval to a maximum of 10 years. Accordingly the staff will defer action on this request pending further discussions with the licensee.

In response to discussions with the staff, the licensee, by letter dated February 10, 1995, has requested a one time surveillance interval extension until start up from refueling outage eight (RFO 8) currently scheduled for fall of 1996 while continuing to discuss their request for a 10 year surveillance interval.

This submittal was identified by the licensee as a cost-beneficial licensing action (CBLA). This means that the proposed action represents a large cost savings to the licensee without a commensurate safety benefit. Although the change does have safety benefit (e.g., occupational dose reduction due to reduced testing), the major benefit to the licensee is economic. GGNS expects cost reductions of at least $10 million over the remaining life of the plant due mostly to reduced surveillance testing. The anticipated total cost savings surpass the staff's threshold of $100,000 established under the CBLA program, and is therefore acceptable to be considered under the CBLA program.

The effect of steam bypass of the suppression pool on primary containment integrity has been evaluated for a spectrum of break sizes. The limiting case (assuming containment sprays and heat sinks are available) results in a maximum allowable leakage path area of 0.90 ft^2. (Maximum leak path areas are expressed in terms of $A/\sqrt{k}$, where A is the flow area of leakage and k is the geometric and friction loss coefficient.) The value $A/\sqrt{k}$ of 0.90 ft^2 is equivalent to a bypass leakage rate of 35,000 scfm at a drywell design pressure of 3.0 psid drywell to containment.

The drywell bypass leakage has been measured periodically since completion of construction in 1982. These surveillances have been carried out 3.0 psid between drywell and containment with a corresponding allowable leakage limit of 3,500 scfm. The results of these tests are summarized below:

Previous Results of GGNS Drywell Bypass Leakage Rate Tests

Test Date	Leak Rate (at 3.0 psig)	Ratio to Design Limit	Calculated $A/\sqrt{k}$
01/82	0611 scfm	1.75%	0.016 ft^2
03/83*	1621 scfm	4.63%	0.042 ft^2
06/84	2599 scfm	7.43%	0.067 ft^2
11/85	2315 scfm	6.61%	0.060 ft^2
11/86(RF01)	1568 scfm	4.48%	0.040 ft^2
12/87(RF02)	1500 scfm	4.28%	0.039 ft^2
04/89(RF03)	1631 scfm	4.66%	0.042 ft^2
11/90(RF04)	1591 scfm	4.55%	0.041 ft^2
05/92(RF05)	0618 scfm	1.77%	0.016 ft^2
11/93(RF06)	0869 scfm	2.48%	0.022 ft^2

* NOTE: The initial test failed due to open drywell penetrations.

Based on the 10 successful test results summarized above which reveal an $A/\sqrt{k}$ that is a small fraction of the allowable limit, the licensee believes that a reduction in testing is warranted.

The staff's concern over decreasing the frequency of performing DBLRTs is that potential sources of steam bypass leakage paths could remain unidentified for an extended period ot time. Potential sources include cracks in the drywell concrete structure, the drywell vacuum breakers, the drywell air locks, and drywell piping penetrations. The licensee's submittal addressed these potential bypass leakage paths, as summarized below.

The preoperational drywell structural integrity test was conducted at 30 psig. The test results indicated that the drywell was not stressed as much as predicted and responded in the elastic stress range. No signs of permanent damage to either the concrete liner or the liner were detected. A measured drywell bypass leakage rate of 3200 scfm was measured for the drywell structure during the preoperational test at a differential pressure of 30 psi with the containment open to the atmosphere and the drywell to suppression pool vents sealed. Subsequent drywell bypass leakage rate tests have been carried out 3.0 psig and have not indicated any significant degradation of the drywell integrity with time. During normal operation the drywell is operated at a nominal pressure of approximately 1.0 psi. Visual inspections of the drywell structure conducted during each refueling outage have not revealed any additional cracks. Therefore, the staff concludes that additional cracking of the drywell structure is not expected due to testing or operation.

Piping penetrations having containment isolation valves do not represent a significant concern with regard to drywell bypass leakage. This is because containment isolation valves are locally leak-tested in accordance with 10 CFR Part 50, Appendix J and the leakage limitations for Appendix J are much lower than that allowed for drywell leakage. However, drywell penetrations not subject to local leak rate testing are of special concern for drywell bypass leakage. Leakage through the post-LOCA vacuum relief valves is minimized by the use of two valves in series that are normally sealed shut by the slightly higher pressure in the drywell. In addition, TSs require that these valves be verified to be in the closed position at least once per 7 days. The drywell vent and purge system has two 6-inch supply isolation valves in series. TSs require that both of these valves be sealed closed during Operational Conditions 1, 2, and 3 and that they be verified to be in the closed position at least once per 31 days. The drywell vent and purge system has two exhaust isolation valves. TSs only permit these valves to be open for a total of five hours per 365 days. The staff finds that these controls provide reasonable assurance of preventing unacceptable bypass leakage flow.

Another potential bypass leakage path is through the drywell personnel air locks. The licensee has proposed that one drywell door shall remain open during the drywell leakage test, such that each drywell door is leak tested during at least every other leakage rate test. While the DBLRT will only test the air lock doors once every 10 years, TS 3/4.6.2.3, "Drywell Air Locks," requires an overall air lock leakage test at $\geq$ 11.5 psig at least once per 18 months. The staff finds that these controls provide reasonable assurance of preventing unacceptable bypass leakage flow.

The licensee has evaluated the risk impact of the proposed changes to determine the magnitude of a postulated increase in post accident releases from the containment. The GGNS individual plant evaluation (IPE) carried out per NRC Generic Letter 88-20 was used to evaluate the potential impact on containment releases that could occur due to longer drywell bypass leakage surveillance intervals. An analysis was conducted to determine the potential risk to the public due to the increased probability that a large increase in drywell bypass leakage could go undetected for an extended period of time. The licensee's analysis estimated that the added risk of radioactivity release from containment was due to excessive drywell leakage was less than 1E-7 conservatively assuming that the probability of excessive drywell leakage is 1E-2 and that the calculated failure probability to function on demand is approximately 1E-3. The estimated frequency for this event is very low and is of the order of magnitude of the low frequency severe accident events considered in the GGNS IPE. The resulting potential release is significantly smaller than those estimated for severe accident sequences of comparable frequency and does not increase the estimated overall plant risk.

In summary, the licensee has provided an extensive and diverse justification to decrease the frequency of performing DBLRTs. The performance of DBLRTs is expensive and adds to the outage critical path. Past DBLRTs performed at the GGNS have consistently demonstrated that the measured drywell bypass leakage is a small fraction of the allowed leakage. The potential bypass leak paths

of most concern, have been addressed by the licensee and reasonable assurance has been provided to prevent them from becoming significant contributors to bypass leakage paths. Finally, a risk analysis adds further support to extend the test interval. Therefore, based on the information provided by the licensee, the staff concludes that it is acceptable to defer the drywell bypass surveillance test scheduled for RFO 7 until startup from RFO 8. A note to that effect is added to the TS. In the meantime, the staff is reviewing the licensee's broader request for a 10-year surveillance interval.

Change No. 7. This change modifies the TS LCO actions and surveillances for the drywell airlock in TS 3/4.6.2.3 to be consistent with the format and content of NUREG-1434.

7c. Existing TS 3.6.2.3 is proposed to be modified with a new allowance. This new Condition B applies solely to an inoperable air lock interlock mechanism. If access into containment is desired, Note 2 permits an individual to be stationed at the air lock and dedicated to assuring that two doors are not open simultaneously and one door is re-locked prior to leaving. This individual thus provides substantially the same level of protection as if the interlock mechanism were operable. The condition further provides for periodic verifications that the air lock door remains locked until the interlock mechanism is returned to operable status and the condition is exited. The staff finds that this alternate required action assures the air lock door is closed to match the assumptions of the accident analyses.

7e. Existing TS 3.6.2.3 is proposed to be modified with a new condition. An Action Note #1 is being added to permit entry through a closed or locked primary containment air lock door for the sole purpose of making repairs. An air lock with an inoperable outer door is fully accessible and the operable inner door maintains the containment operable. This Note would only apply to repairs made to an inoperable inner door of the air lock. Without this allowance, the unrepaired door could prevent the overall air lock test from being performed and thus result in a plant shutdown from the inability to demonstrate the air lock operable. Additionally, it is the staff's preference to keep both doors operable in each air lock as an improvement on safety over just one operable door locked closed. It is possible to gain access to the inoperable inner door by entering the containment from the other air lock. If this is not practical due to the length of travel distance or exposure considerations, then this Note would be utilized. This would result in the momentary loss of the primary containment boundary as the outer door is opened for entry. The staff finds this to be acceptable due to the low probability of an event occurring that could pressurize the primary containment during the short time in which the containment boundary is compromised.

7f. Existing TS 3.6.2.3 is proposed to be modified with a new condition

that allows entry and exit through a closed and/or locked airlock door (for reasons other than repairs) for a limited period of time (i.e., 7 days). Entry and exit through the primary containment air lock during normal operation is necessary to perform required surveillance, maintenance and inspections as well as allowing routine access for operational considerations such as chemistry sampling, reactor water cleanup system operations, refueling preparations, etc. If both air locks become inoperable and access is not allowed, a plant shutdown would be forced in a short period of time due to failure to attend to these required activities. This Condition A, Note #2 is added to allow entry for reasons other than repairs, under strict administrative controls, which are detailed in the Bases, for a period of time not to exceed seven days. In this one-time seven day period, an air lock must be returned to operable status or the forced shutdown must occur. The temporary loss of the containment boundary for brief times during this seven day period is judged by the staff to be acceptable. The risk associated with an event occurring during the brief period of time (not to exceed seven days) is significantly less than the risk associated with a plant shutdown concurrent with plant equipment that may not be in a satisfactory operational condition.

7h. The note "The provisions of Specification 4.0.2 are not applicable" in SR 4.6.2.2.c is proposed to be deleted. This deletion is in conformance with the provisions of NUREG-1434 that states that "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency,---." The provisions of NUREG-1434 further state that "For Frequencies specified as "once," the above interval extension does not apply." In NUREG-1434, the surveillance requirement for verifying drywell bypass leakage, SR 3.6.5.1.1, does not include "once" in its surveillance frequency. The staff finds that the application of a 25% allowance to this 18 month surveillance interval is acceptable to assure adequate flexibility to accommodate extended operating cycles that may be needed to support unexpected system power needs.

The BWR/6 drywell air lock is typically tested similar to Primary containment air locks. However, the drywell air lock is not a direct leakage path from primary containment and therefore Appendix J test requirements do not necessarily apply. Furthermore, its use is limited during operation due to radiation and temperature in the BWR/6 drywell. Since sufficient confidence in its sealing capability is assured via other specified surveillances, it is justified to allow performance of this test at refueling-outage intervals.

7i. Existing TS 4.6.2.3.b.1 requires an overall airlock leakage test every six months. The drywell air lock is typically tested similar to the primary containment air locks; however, the drywell air lock

is not a direct leakage path from the primary containment and, therefore, Appendix J requirements do not apply. In addition, the drywell airlock does not experience frequent usage due to the radiation and temperature in the drywell. The improved TS interval requirement of a barrel test has been extended to 18 months. The staff finds that the verification of the seal leakage rate after each use of the drywell is sufficient to assure sealing capability and thus justifies the scheduling of this test at refueling/outage intervals.

7k. It is proposed that the drywell airlock door interlock operability surveillance not be required to be performed unless the airlock doors are to be opened for a drywell entry. Existing TS 4.6.2.3.c requires verification, once every 18 months, that only one door in each air lock can be opened at a time. A note to improved TS SR 3.6.1.2.2 proposes that this surveillance not be required to be performed unless the air lock doors are to be opened for a containment entry. Without this exception, the air lock doors would be required to be opened solely to perform this interlock test. This scenario would then also require the door seal test be performed within the next 72 hours creating unnecessary containment entries and requiring manpower for testing. In the event the plant is utilizing one air lock for entries and maintaining one air lock idle, this surveillance would impose an excessive testing requirement.

The above less restrictive requirements have been reviewed by the staff and have been found to be acceptable, because they do not present a significant safety question in the operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding

(58 FR 64607). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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