September 6, 2002

Mr. William A. Eaton
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION, ISSUANCE OF AMENDMENT RE: REACTOR CAVITY POOL DRAINDOWN (TAC NO. MB4260)

Dear Mr. Eaton:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 154 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 25, 2002, as supplemented by letters dated August 16 and August 22, 2002.

This amendment adds a new TS 3.10.9 "Suppression Pool Makeup-MODE 3," to allow installation of reactor cavity gate 2 in the Upper Containment Pool (UCP) and draining the reactor cavity pool portion of the UCP while still in MODE 3. The amendment also modifies the applicability of the UCP gates Surveillance Requirement (TS 3.6.2.4, "Suppression Pool Makeup (SPMU) System,") to allow installation of UCP gates in MODES 1, 2, and 3.

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

Sincerely,

/RA/

David Jaffe, Sr. Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
1. Amendment No. 154 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

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* SE input provided - no major changes made.
**See previous concurrences

OFFICE	PDIV-1/PM	PDIV-1/LA	SPLB/SC**	SPSB/SC**	IEHB/SC*	OGC**	PDVI-1/SC
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DATE	9/6/02	9/6/02	9/3/02	9/4/02	8/30/02	09/06/02	9/6/02

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ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

ENTERGY MISSISSIPPI, INC.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. NPF-29

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

   A. The application for amendment filed by Entergy Operations, Inc. (the licensee) dated February 25, 2002, as supplemented by letters dated August 16 and August 22, 2002, 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. 154, 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. The license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 6, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.6-34	3.6-34
---	3.10-23
---	3.10-24
---	3.10-25
---	3.10-26

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154 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 application dated February 25, 2002, as supplemented by letters dated August 16 and August 22, 2002, (References 7.1, 7.2, and 7.3, respectively), Entergy Operations, Inc., et. al. (Entergy or the licensee) submitted a request for changes to the Grand Gulf Nuclear Station, Unit 1 (GGNS or Grand Gulf), Technical Specifications (TSs).

References 7.2 and 7.3 provided clarifying information that did not change the scope of the original *Federal Register* notice (67 FR 21289, published April 30, 2002), or the original no significant hazards consideration determination.

The proposed amendment would add a new TS 3.10.9 "Suppression Pool Makeup-MODE 3," to allow installation of Reactor Cavity Gate 2 in the Upper Containment Pool (UCP) and draining the reactor cavity pool portion of the UCP while still in MODE 3. The amendment also modifies the applicability of the UCP gates Surveillance Requirement (SR) TS 3.6.2.4, "Suppression Pool Makeup (SPMU) System," to allow installation of UCP gates in MODES 1, 2, and 3. Installing the gates partitions the different parts of the upper containment pool and makes the draining of only the reactor cavity portion feasible.

2.0 REGULATORY EVALUATION

2.1 Description of the Upper Containment Pool and its Safety Function

Grand Gulf is a boiling water reactor (BWR/6) with a Mark III containment. The containment building encloses the drywell. The drywell is a cylindrical, reinforced concrete structure with a removable head. The reactor cavity portion of the UCP lies above the drywell head. The drywell encloses the reactor vessel and the reactor coolant system. It is designed to withstand the pressure and temperature of the steam generated by a reactor coolant system pipe rupture, and channel the steam to the suppression pool via horizontal vents located in the drywell wall. The suppression pool contains a large volume of water which rapidly condenses steam directed to it. Initially, the air in the drywell is forced into the suppression pool by the steam discharging from the postulated break and pressurizes the containment. The containment's design pressure is 15 pounds per square inch, gauge (psig).

The drywell provides the structural support for the UCP. The UCP provides several functions: (1) radiation shielding when the reactor is in operation; (2) storage space for the dryer, separator, and fuel assemblies during refueling; (3) an area for fuel transfer during refueling; (4) storage for control blade guides, new blades, refueling tools, and other irradiated and unirradiated components (fuel assemblies are not stored in the UCP during operation); and (5) a large volume of water for the SPMU after a postulated loss-of-coolant accident (LOCA).

Figure 1 of Reference 7.1, Attachment 1, illustrates the regions of the Grand Gulf UCP: (1) the fuel storage pool, (2) the fuel transfer pool, (3) the reactor cavity pool, and (4) the separator pool.

The UCP provides water to the suppression pool following a LOCA by means of the SPMU system described in the GGNS Updated Final Safety Analysis Report (UFSAR) (Reference 7.4), Section 6.2.7. The SPMU system consists of two 100 percent capacity lines, each of which directs a portion of the UCP water to the suppression pool by gravity when the operator opens two normally closed valves in series in each line. The operator opens these valves in response to a low-low suppression pool water level signal. The SPMU system will also dump water to the suppression pool automatically upon a signal from a timer set for 30 minutes after the LOCA begins.

The location of the SPMU system pipe inlet limits the amount of UCP water which can be dumped to the suppression pool.

In order to ensure the proper amount of water is dumped to the suppression pool, the current TS SR 3.6.2.4.4 requires that all UCP gates, which separate the various volumes of the UCP, must be removed when in MODEs 1, 2, and 3.

The water supplied by the SPMU system, together with the water inventory in the suppression pool, is sufficient for all safety-related functions of the suppression pool. These safety-related functions include: (1) providing the emergency core cooling system (ECCS) with a source of water for injection into the vessel following a LOCA, (2) providing a heat sink for the decay and sensible heat released during reactor blowdown from the safety/relief valves (S/RVs) or from a LOCA, (3) providing adequate net positive suction head (NPSH) to the ECCS pumps, (4) condensing steam discharged from the reactor core isolation cooling (RCIC) system turbine, (5) providing a long-term heat sink for cooldown of the reactor, and (6) maintaining structural loads on the drywell and containment structures within acceptable limits.

The Grand Gulf TSs specify limits on the minimum and maximum suppression pool level. The minimum suppression pool water level limit ensures adequate coverage of the horizontal vents during the initial portion of the LOCA. This ensures that steam discharged from the S/RV quenchers, main vents, and the RCIC turbine exhaust lines is completely condensed. The ECCS takes suction from the suppression pool. The suppression pool water is injected into the reactor vessel and spills out of the break. This water forms a pool in the bottom of the drywell inside the weir wall. This pool is referred to as the drywell pool. The water in the drywell pool is not available to the suppression pool until the water level rises sufficiently to overflow the drywell weir wall; this overflow returns to the suppression pool. The water inside and below the top of the weir wall remains unavailable, as well as water entrapped in other volumes. This entrapped water reduces the volume of water in the suppression pool; this is referred to as suppression pool drawdown. The sizing of the UCP provides sufficient water to the suppression pool to maintain the minimum suppression pool level two feet above the top row of vents, considering the entrapped water which cannot return to the suppression pool.

The entrapped volumes considered in the current analysis are: (1) the free volume inside and below the top of the drywell weir wall; (2) the added water volume needed to fill the vessel from the level at normal power operation to a post-accident complete fill of the vessel, including the top dome; (3) the volume in the steam lines out to the first main steam isolation valve (MSIV) for three lines and out to the second MSIV in the fourth line (assuming a single failure); and (4) the containment spray hold-up on equipment and structural surfaces.

The maximum suppression pool water level limit ensures that clearing loads from Safety/Relief Valve(S/RV) discharges and suppression pool swell loads will not be excessive. The maximum level also ensures adequate freeboard so that an inadvertent dump of water from the UCP will not overflow the weir wall into the drywell.

2.2 Regulatory Requirements and Guidance

Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases," requires that structures, systems, and components important to safety (such as the containment and the suppression pool) shall be appropriately protected against dynamic effects. Standard Review Plan (SRP) Section 6.2.1.1.c (Reference 7.5) specifies that this includes suppression pool dynamic effects during a LOCA or following the actuation of one or more reactor coolant system S/RVs. The licensee's proposal affects the water level in the suppression pool and, therefore, affects the hydrodynamic loads on the containment structure, including the drywell and the suppression pool.

GDC 16, "Containment Design," requires the containment to be a "leak tight barrier" and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The initial water inventory in the suppression pool and the additional water supplied by the SPMU system are important for compliance with the requirements of GDC 16.

GDC 38, "Containment Heat Removal," requires that the containment heat removal system remove heat from the reactor containment following a LOCA so that the containment pressure and temperature following a LOCA will be maintained at acceptably low levels. The initial water inventory in the suppression pool and the additional water supplied by the SPMU system are important for compliance with the requirements of GDC 38.

GDC 50, "Containment Design Basis," requires that the reactor containment structure and its internal compartments accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. The margin must reflect consideration of the conservatism of the calculation model and input parameters. The initial water inventory in the suppression pool and the additional water supplied by the SPMU system are important for compliance with the requirements of GDC 50. In addition, to justify the proposed TS changes, the licensee is proposing to use a computer code that has not been previously applied as part of the Grand Gulf licensing basis. This analytical method is different from that specified as acceptable to the U.S. Nuclear Regulatory Commission (NRC or the Commission) in Reference 7.5, Sub-section II.

Reference 7.4, Section 6.2.7.1, lists the twelve design bases for the SPMU system. The staff's evaluation of the compliance of the proposed changes with these design bases is discussed in

Section 3.0 of this Safety Evaluation (SE). The effect of the proposed changes on the design bases are listed below:

a. The suppression pool makeup system is redundant with two 100 percent capacity lines. The redundant lines are physically separated and the electrical power and control is separated into two divisions in accordance with IEEE [Institute of Electrical and electronics Engineers] St[andar]d 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," 1971.

The physical design of the system will not change and, therefore, this design basis will not change.

b. The suppression pool makeup system is Safety Class 2, Seismic Category 1, and Quality Group B.

The physical design of the system will not change and therefore, this design basis will not change.

c. The minimum long term post-accident suppression pool water coverage over the top of the uppermost drywell vents is two feet.

This design basis has not changed; it is used by the licensee to demonstrate that the proposed TSs changes are acceptable.

d. The minimum normal operation low water level suppression pool height above the top drywell vent centerline is 7 feet 1/12 inches.

This criterion has changed as a result of proposed changes. This is discussed in Section 3.0 of this SE.

e. The maximum normal operation high water level suppression pool height above the top drywell vent centerline is 7 feet 5-3/4 inches.

This criterion has changed as a result of proposed changes. This is discussed in Section 3.0 of this SE.

f. The suppression pool volume, between normal low water level and the minimum post-accident pool water level, plus the makeup volume from the upper pool, is adequate to supply all possible post-accident entrapment volumes for suppression pool water.

The licensee has revised the amount of water assumed to be in the entrapped volumes. In addition, TS Figure 3.10.9-1, “Upper Containment and Suppression Pool Levels,” changes the required amount of water in the suppression pool as a function of the water in the UCP. In consideration of the large-break LOCA with maximum drywell bypass, water makeup from an external source of water to the suppression pool is needed, as well as water from the UCP. This is discussed in Section 3.0 of this SE.

g. The post-accident entrapment volumes causing suppression pool level drawdown include:

1. The free volume inside and below the top of the drywell weir wall,

2. The added water volume needed to fill the vessel from a condition of normal power operation to a post-accident complete fill of the vessel, including the top dome,

3. Volume in the steam lines out to the first MSIV for three lines and out to the second MSIV on one line (single failure), and

4. An allowance for containment spray hold-up on equipment and structural surfaces.

The proposed changes to the technical specifications revise item 1. by reducing this free volume by the volume of equipment and structures located in this part of the drywell, and revise item 2. by taking credit for operator action to limit the reactor vessel water level to the narrow range high water level (Level 8) in accordance with the emergency operating procedures. This is evaluated in Section 3.0 of this SE.

h. No credit for feedwater or high pressure core spray injection from condensate is taken in calculating minimum post-accident suppression pool water level.

This design basis is not changed.

i. The minimum freeboard distance from the suppression pool high water level to the top of the weir wall is adequate to store the upper containment pool makeup volume and minimize the probability of flooding into the drywell over the weir wall in case of an inadvertent upper pool dump.

This design basis has not changed and is evaluated in Section 3.0 of this safety evaluation report.

j. The minimum normal operation suppression pool volume at the low water level is adequate to act as a short term energy sink without credit for water in the upper containment pool. The short term energy load on the pool consists of hot standby operation for 1-1/2 hours followed by a LOCA.

This design basis has not changed.

k. The long-term containment pressure and suppression pool temperature takes credit for the volume added post-accident from the upper containment pool.

This design basis has not changed.

l. The upper containment pool makeup volume dumps within a time period so that the minimum vent coverage of 2 feet above the top edge of the top vent is maintained, considering: (1) maximum runout flow of all five ECCS pumps, (2) the initial suppression pool water level is at low-low water level (LLWL), and (3) inventory addition to the drywell is through the postulated pipe break.

This design basis has not changed but was re-evaluated for the new suppression pool and upper containment pool levels.

The staff finds that the licensee, in Sections 3.0, 4.0, and 5.1 of Reference 7.1, identified the applicable regulatory requirements. The regulatory requirements and regulatory guidance which the staff considered in reviewing the requested action include::

- 10 CFR Part 50, Section 50.34(f)(2)(ii) and (iii);
- 10 CFR Part 50, Section 50.67;
- 10CFR Part 50, Appendix A, GDC-4;
- 10CFR Part 50, Appendix A, GDC-16;
- 10CFR Part 50, Appendix A, GDC-38;
- 10CFR Part 50, Appendix A, GDC-50;
- 10 CFR 50.90, regarding changes to TSs;
- 10 CFR 50.92, regarding No Significant Hazards Consideration;
- 10 CFR 50.120;
- NUREG-0800 (Reference 7.5), Chapter 18, "Human Factor Engineering;"
- American National Standards Institute (ANSI)/American Nuclear Society (ANS) 58.8, "Time Response Design Criteria for Safety Related Operator Actions," 1994;
- NRC Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," 1997; and
- Regulatory Guide 1.183 (Reference 7.6).

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses, in support of its proposed license amendment, which are described in Sections 3.0, 4.0, and 5.0 of References 7.1, 7.2, and 7.3. The detailed evaluation below will support the conclusion 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.

3.1 Hydrodynamic Load Consideration

The licensee's proposed addition of Limiting Condition for Operation (LCO) 3.10.9 permits a higher suppression pool high water level (HWL) in MODE 3 than currently allowed by LCO 3.6.2.2. TS Figure 3.10.9-1, for the case of a drained reactor cavity pool in MODE 3, allows the suppression pool HWL to increase to 20 ft.-6 in. which is an increase of 1 ft.-8-1/4 in. above the current HWL permitted by LCO 3.6.2.2. For consideration of hydrodynamic loads, Reference 7.1 assumed the suppression pool HWL to be 20 ft.-7 in.

The licensee evaluated the effects of increasing the suppression pool HWL limit on the hydrodynamic loads due to either a LOCA or a S/RV actuation.

NUREG-0978 (Reference 7.7) specifies the load definitions to be used for evaluating hydrodynamic loads for Mark III containments. The specific loads for Grand Gulf are discussed in Reference 7.4, Appendices 6A and 6B. Reference 7.1 states that the licensee considered each of these containment loads.

The hydrodynamic loads considered are: water jet loads, LOCA air bubble loads, pool swell drag and impact loads, fallback loads, froth impingement and drag loads, condensation oscillation and chugging loads, and drywell depressurization loads.

In addition to these loads, the containment is also subject to drywell depressurization loads, assuming the operator does not throttle the ECCS pumps before 600 seconds and the relatively cold water spilling out of the break condenses steam, which results in the drywell pressure decreasing below that of the containment so that the drywell structure is subject to compressive loads.

The licensee also considered the effect of the level change on the ECCS suction strainers.

The Table below summarizes the results of the licensee's analyses:

Effect of Increasing the Suppression Pool HWL in MODE 3

Load	Effect
Water Jet Loads	These loads are small compared with other loads
LOCA Air Bubble Loads	The lower reactor pressure results in a decrease in these loads compared with the case of a LOCA at full reactor pressure
Pool Swell Drag and Impact Loads	Increase in pool level increases the bubble pressure. However, with the decrease in reactor pressure, the drywell pressure driving the bubble is smaller. The licensee's GOTHIC calculations show that at the lower reactor pressure, the top vent passes almost all the flow. Reference 7.7 shows that loads are greatly reduced when only the top vent clears.
Fallback Loads	Since the pool swell is bounded by the design basis accident (DBA), the fallback impact loads will be less for a low pressure LOCA.
Froth Impingement and Drag Loads	These loads are bounded by the DBA case
Condensation Oscillation and Chugging Loads	Condensation oscillations are independent of vent submergence and, therefore, not increased by these changes. Resolution of Humphrey Concern 19.1 (Reference 7.8) demonstrated that the margins inherent in the chugging loads are adequate to accommodate increased vent submergence up to 4.5 ft., which is less than allowed by the licensee's proposed changes. The major hydrodynamic loads on the ECCS suction strainers are due to chugging and are bounded by the MODE 1 case.
S/RV Discharge Loads	Increased suppression pool level will increase these loads. However, resolution of Issue 19.1 of Reference 7.8 has shown these loads to be acceptable for suppression levels greater than allowed by this change to the Grand Gulf TSs. In addition, at lower reactor coolant system pressure, the loads will be much lower than those from an S/RV discharge from full reactor coolant system pressure.
Drywell Depressurization Loads	A MODE 3 LOCA results in a lower containment pressure, which results in lower drywell depressurization loads.

For the reasons set forth in the Table, the staff finds the licensee's conclusion, that the effect on hydrodynamic loads imparted with 20 ft.-7 in. suppression pool level and the reactor pressure

$\leq$ 235 psig are bounded by those from a DBA with the suppression pool at the current HWL limit, acceptable.

3.2 <u>NPSH of the Emergency Core Cooling System Pumps</u>

Appendix 6E of Reference 7.4 contains the licensee's calculations of NPSH for the ECCS pumps. These analyses were performed for a suppression pool minimum drawdown water level of 14.5 ft. above the bottom of the suppression pool (which is two feet above the top vent). These analyses show that the ECCS pumps have adequate available NPSH and also that other deleterious pump suction phenomena such as vortexing (air entrainment) or flashing will not occur. Since the design basis of a minimum suppression pool level of 14.5 ft. is not changed by this proposed amendment, the licensee's calculations remain valid and are acceptable.

3.3 <u>Long Term Heat Sink</u>

The suppression pool provides the long-term heat sink for the decay and sensible heat released following a LOCA. The suppression pool cooling system transfers the heat from the suppression pool to the ultimate heat sink to keep the suppression pool within its design basis.

The long-term suppression pool volume is reduced by 570 ft^3 due to the proposed changes (Reference 7.1). Reference 7.2 shows how this quantity was determined. This reduction in water inventory results in an increase in the suppression pool temperature, and the licensee examined the consequent effect of increased pool temperature on the containment pressure and temperature. The containment air is assumed to be in thermal equilibrium with the suppression pool and, therefore, containment pressure increases with suppression pool temperature.

The licensee had previously responded to concerns relating to the suppression pool listed in Reference 7.8. The responses were evaluated by the NRC. The NRC's evaluation is given in supplements to the Grand Gulf Safety Evaluation Report (Reference 7.9) and in a March 23, 1987, letter from Lester L. Kintner, USNRC, to Oliver D. Kingsley, Jr., System Energy Resources, Inc. (Reference 7.10).

The impact of long-term reduced suppression pool inventory was addressed in Item 4.1 of Reference 7.9. This issue dealt with the isolation of the drywell volume from the suppression pool and the resulting effect on suppression pool temperature. Resolution of this issue demonstrated that a large reduction in suppression pool water inventory will not significantly affect the long-term suppression pool temperature. Reference 7.2 concludes, on the basis of this analysis, that a reduction in suppression pool water volume of 570 ft^3 will be inconsequential; for the same reasons, the NRC staff has reached the same conclusion.

One of the assumptions listed in Reference 7.4, Section 6.2.1.1.3.3.1.3, is that the suppression pool is the only heat sink available in the containment system. This proposed amendment revises this assumption slightly since make-up to the suppression pool is required for large-break LOCAs with drywell bypass. See Section 3.4 of this SE.

3.4 Drywell Bypass

3.4.1 Description of Drywell Bypass Scenario

Grand Gulf's Mark III containment is designed so that any steam released from the reactor coolant system into the drywell is directed to and condensed in the suppression pool, and will not, therefore, contribute to increasing the containment pressure. However, the design basis for the containment assumes that a limited amount of the steam bypasses the suppression pool and pressurizes the containment. This is termed drywell bypass leakage.

Section 6.2.1.1.5 of Reference 7.4 discusses LOCAs with drywell bypass leakage. The allowable leakage is defined in Reference 7.4, Section 6.2.1.1.5.2, as the amount of steam which could bypass the suppression pool without exceeding the design containment pressure of 15 psig. Reference 7.4 analyses assume operation in MODE 1 at full reactor pressure prior to the LOCA. The most limiting break is a small break which (1) does not actuate the automatic depressurization system to depressurize the reactor coolant system, and (2) maintains the level in the suppression pool annulus above the top of the top row of horizontal vents. This maximizes the pressurization of the drywell and, therefore, maximizes the drywell bypass leakage. It is assumed that the operators depressurize the reactor coolant system within a conservatively long six hours.

The bypass leakage is described in terms of the parameter $A/\sqrt{K}$, where A is the flow area of the leakage path and K is a measure of the flow resistance of the leakage path. The limiting value of $A/\sqrt{K}$ for the allowable drywell bypass leakage from MODE 1, taking credit for the containment spray and heat sinks, is 0.9 ft^2 (Reference 7.4, Section 6.2.1.1.5.5).

The licensee examined the effect of drywell bypass leakage on the proposed TS changes and found that increasing the suppression pool level increases the pressure in the drywell required to clear the top vent and, therefore, potentially could increase the drywell bypass leakage. The evaluation of small-break and large-break LOCAs with drywell bypass leakage is presented in Sections 3.4.2 and 3.4.3, herein.

3.4.2 Small-Break Loss-of-Coolant Accident with Steam Bypass of the Suppression Pool in MODE 3

The licensee used the GOTHIC computer code to determine the impact of raising the suppression pool level on drywell bypass leakage with the reactor in MODE 3 and the reactor pressure equal to 235 psig. The drywell bypass leakage used was the existing design basis value of $A/\sqrt{K}$ = 0.9 ft^2. This is the maximum licensed drywell bypass leakage and is limiting for all break sizes. Measured values of $A/\sqrt{K}$ from the Grand Gulf TS surveillance tests are much less than this value (Reference 7.11). The licensee stated (Reference 7.1) that the size of the break assumed is approximately 0.07 ft^2.

The licensee's calculations show that containment spray would initiate for this event. This is not a concern because the drywell pool does not form or forms only minimally. The drywell pool volume is approximately twice the spray hold up volume. Therefore, adequate water will remain available for this event.

3.4.3 Large-Break Loss-of-Coolant Accident with Steam Bypass of the Suppression Pool in MODE 3

Reference 7.9, Issue 5.1, asserted that the worst case of drywell bypass leakage had been established as a small-break accident, but an intermediate break accident will actually produce the most limiting drywell bypass leakage prior to initiation of the containment sprays. The NRC staff addressed this issue in Reference 7.9. The NRC stated that the concern was adequately addressed by the licensee, since NRC requires Mark III owners to consider the entire spectrum of break sizes in their analyses of drywell bypass leakage. These matters are discussed in detail below.

Since the licensee has evaluated the drywell bypass leakage for a different condition and containment configuration, namely MODE 3 with increased suppression pool level, decreased inventory in the UCP, and the reactor cavity portion of the UCP drained, the licensee re-evaluated this event for a spectrum of break sizes.

The original analysis for MODE 1 conditions and the current requirements of suppression pool and UCP level assume a drywell bypass leakage of $A/\sqrt{K}$ equal to 0.9 ft^2, one loop of containment spray initiated at 13 minutes, credit for structural heat sinks, and reactor vessel level control in accordance with the emergency operating procedures (EOPs) to minimize the overflow through the break. These calculations determined the most limiting break to be a 2.5 ft^2 main steam line break and predicted the peak containment pressure to be 14.5 psig prior to containment spray initiation. The licensee repeated the analysis for MODE 3 conditions, with the suppression pool level increased and a reduced pressure of 235 psig. The GOTHIC computer code was used. The peak containment pressure prior to spray initiation is 12.23 psig; less than the MODE 1 pressure. Thus, the MODE 1 drywell bypass leakage calculation is limiting.

The licensee has determined that containment spray actuation, while in the proposed LCO 3.10.9, would result in an additional entrapped volume in the drained reactor cavity pool equal to about 30 percent of the drywell pool volume. Therefore, for the large main steam line break drywell bypass leakage events, the volume of water trapped in the drywell pool must be limited to 70 percent of the total drywell pool volume to ensure adequate suppression pool inventory. The licensee states in Reference 7.2 that this can be done if the operator follows the EOPs and controls the reactor water level to Level 8 to limit spillage from the break into the drywell pool. The operator’s control of reactor vessel level to Level 8 is discussed in Section 3.9 of this SE.

The large-break LOCA from MODE 3 with drywell bypass is the only event which requires makeup from an external source to the suppression pool. The licensee determined that the most limiting break size for drywell bypass leakage response time is a 3.54 ft^2 main steam line break with maximum ECCS. This case maximizes spillage from the break and the rate of drywell pool formation.

Reference 7.1 provides the following description of the timing of SPMU operation for the worst case.

> A GOTHIC analysis of the 3.54 ft^2 special bypass leakage capability event was performed assuming that operators must control reactor vessel [level] within 7.5 minutes after accident initiation. Once the reactor vessel level is reduced below the break

> elevation, continued steaming will slowly deplete the suppression pool inventory as the steam from the break condenses and is entrapped in the Drywell pool. Assuming that two loops of CS [containment spray] start at 10.75 minutes and run continuously and all steam from the break condenses in the drywell pool, external makeup flow to the Containment will be required no earlier than 6 hours after accident initiation. If operators control vessel level within 10 minutes, external makeup flow will be required within 1 hour 23 minutes after accident initiation.

The licensee further states (Reference 7.1) that:

> Since this event assumes maximum ECCS, all division power is available and, considering the low pressure MODE 3 conditions, the 10 minute operator response time is reasonable.

Makeup to the suppression pool is available from several sources, both safety related and not safety related, including the main feedwater system, the condensate storage tank, and the standby service water system crosstie.

The licensee, in Reference 7.2, concludes a discussion of operator actions by stating:

> Considering the conservative assumptions used in the bounding MSLB [main steam line break] bypass leakage analyses (e.g., continuous operation of two spray loops), and the above considerations for operator action [close attention to reactor conditions since the plant is being shutdown, safety-related source available, available procedures, and training], a 1 hour 23 minute response time for this action [makeup to the suppression pool from an external source] is reasonable.

Based on the above, the staff finds that, for this case, the makeup to the suppression pool is acceptable, including the available time for operator response and the availability of water sources.

3.5 <u>Drywell Freeboard</u>

To prevent drywell flooding during normal operation and transients, the weir wall height is designed to prevent overflow into the drywell for the case of an inadvertent actuation of the SPMU system with the maximum amount of water available. Reference 7.1 states that with the suppression pool at the HWL permitted in MODE 3 (21-1/4 in. higher than the current HWL limit) the freeboard is reduced to 3 ft.-9 in. However, the licensee has determined that an UCP dump at the most limiting point in the cavity-drained evolution raises the suppression pool level to approximately 2 ft.-2 in. which leaves 1 ft.-7 in. of weir wall height available to prevent overflow into the drywell. The licensee further states that LCO 3.6.5.4 limits the drywell-to-containment differential pressure to -0.26 pounds per square inch, differential (psid) and prevents overflow into the drywell even with the higher starting pool water elevation and an upper pool dump. This satisfies Reference 7.4, Section 6.2.7.1.i. Since the design basis continues to be satisfied, the staff finds the proposed changes acceptable with respect to drywell freeboard.

3.6 Dump Time

Section 4.3.8 of Reference 7.1 discusses the SPMU system dump time. The dump time criterion ensures that the water from the UCP reaches the suppression pool fast enough so that the suppression pool level will not fall below the minimum vent coverage of 2 ft. above the top of the top horizontal vent. This satisfies Section 6.2.7.1.l of Reference 7.4. The dump time accounts for the reduction in the suppression pool water level due to operation of all five ECCS pumps, and the opening time of the SPMU system inlet valves. Section 4.3.8 of Reference 7.1 states that the “dump time criterion is met” for the proposed TS changes. Since the design basis continues to be satisfied, the staff finds the proposed changes acceptable with respect to SPMU system dump time.

3.7 Dose Analysis

The licensee performed calculations of off-site and control room dose as part of the justification for the proposed changes to the TSs. These are discussed in Reference 7.1. Additional information was provided in Reference 7.3.

The original design basis LOCA dose analysis credits containment spray for fission product removal in the containment. Since the licensee’s GOTHIC calculations show that the containment spray will not actuate automatically in the event of a large break LOCA from MODE 3 conditions and emergency operating procedures for a LOCA will not lead the operator to manually actuate the containment sprays, the licensee did not credit containment sprays for the MODE 3 dose analyses.

The licensee performed a dose analysis for the case of a LOCA with the initial plant condition of MODE 3 with the reactor pressure less than 235 psig and the reactor subcritical for greater than three hours. The licensee states (Reference 7.3) that these analyses were performed with the same methodology as used in the Grand Gulf alternative source term analyses, which was approved and accepted by the staff in License Amendment 145 to the Grand Gulf Facility Operating License, dated March 14, 2001.

Containment spray results in increased mixing between the unsprayed and the sprayed regions. In the absence of credit for containment spray, the licensee credited natural deposition in the containment. Natural deposition has removal rates that are significantly less than those due to containment spray. The reactor cavity portion of the UCP cannot be drained until the reactor has been subcritical for greater than three hours. This results in a significant reduction in the short-lived isotopes before a potential release.

Since the LOCA of interest occurs from MODE 3 conditions with the reactor pressure less than 235 psig, the licensee adjusted the containment leakage rate to reflect the reduced pressure. However, the licensee assumed the containment leakage rate to be constant for 30 days rather than being reduced by 50 percent after 24 hours in accordance with the guidance of Reference 7.6.

The licensee considered two single failures and concluded that the worst single failure is a failure of a MSIV to close. This is the same worst single failure as for a LOCA from MODE 1.

The licensee states that the MODE 3 LOCA dose results meet the acceptance criteria of 10 CFR 50.67 for the exclusion area boundary, low population zone, and control room. Based on the above considerations, the staff finds the licensee's analysis sufficiently conservative and acceptable.

3.8 Human Factors Considerations

3.8.1 Overview

This evaluation addresses operator performance effects resulting from permitting the draining of the reactor cavity portion of the UCP while in MODE 3. The evaluation includes changes to operator actions, and changes to procedures and training. Operator performance dealing with control of vessel level and manual spray actuation are also discussed in Section 3.10 of this SE.

The evaluation is based on a contrast of the response time available prior to manual action by the operator in the control room between situation 1 and situation 2. Situation 1 is a LOCA in MODE 3 with water in the UCP at the currently required water level limit of 23 ft.-3 in. Situation 2 is a LOCA in MODE 3 with the reactor cavity portion of the UCP drained.

Reference 7.1 assumes that the operator controls the post-LOCA reflood of the reactor vessel between the top of the active fuel (TAF) and Level 8 in accordance with the EOPs. Control of the reactor vessel level within this band is necessary to ensure the availability of the suppression pool water assumed in the safety analysis. Maintaining the reactor vessel level within this band ensures that the post-accident suppression pool water level would be maintained at or above 14 ft.-6 in. This is the level required to maintain the minimum suppression pool water level two feet above the top of the highest horizontal vent.

Reference 7.2 specified that for Situation 2, operator action to control reactor vessel water level to between the TAF and Level 8 in accordance with the EOPs is necessary.

The licensee also analyzed a large-break LOCA with design basis drywell bypass leakage and the reactor cavity of the UCP drained. For the large-break LOCA in MODE 3 with drywell bypass, the licensee's calculations show that makeup from an external water source to the suppression pool is required. The limiting break with respect to the timing of the addition of water from an external source to the suppression pool is a 3.54 ft^2 main steam line break. Containment spray actuation is expected and a large quantity of water will become entrapped in both the drained reactor cavity portion of the UCP from the containment spray or in the drywell pool by means of condensed water from the steam flow discharging from the break. The results for maximum ECCS flow show that if the operator controls reactor vessel water level in 10 minutes, external makeup to the suppression pool will not be required before 1 hour 23 minutes.

Sections 3.8.2 through 3.8.4, herein, provide further discussion and staff conclusions on these topics.

3.8.2 Operator Actions

Reference 7.2 states that there are no new operator actions required as a result of a LOCA from MODEs 1, 2, and 3 with the gates installed, or in MODE 3 with the UCP drained. Also, the level of difficulty associated with the manual actions is not increased by the proposed TS changes.

The licensee states (Reference 7.2) that:

> The operators would respond to control level in accordance with the EOPs... the action to control reactor vessel water level below Level 8 is already contained in the EOPs. Current guidance has the operator tak[ing] manual action to control reactor vessel water level between Level 3 (Low Level Scram Setpoint) and Level 8 (High Level Scram Setpoint). The Level 8 Setpoint is approximately 5 feet below the MSLs [main steam lines].
>
> Industry operating experience concerning overfill situations has been incorporated into operator training and is an essential part of their training. Operators are trained and graded on their ability to take prompt actions to limit filling the reactor vessel above the Level 8 limit. This requirement has long been an integral part of their training. Thus, there are no new operator actions as a result of this submittal.

The NRC staff concludes that the licensee has adequately addressed the issue of the Operator Actions with regard to controlling the reactor vessel water level between Level 3 (Low Level Scram Setpoint) and Level 8 (High Level Scram Setpoint), and to taking prompt action to limit filling the reactor vessel above the Level 8 limit.

3.8.3 Procedures

Regarding procedures, Reference 7.2 states:

> Due to the symptomatic nature of the EOPs, the guidance for maintaining suppression pool level simply tells the operator to raise or lower level depending on the circumstances. The action to commence raising level is taken as soon as level decreases below the TS low limit. A list of systems, including the associated system operating instruction (SOI) is provided in the EOPs to assist the operator in performing the actions necessary to raise level in the suppression pool. This is a fairly simple task and for the situation in question, there are four different methods identified.
>
> Due to the symptomatic approach of the EOPs, operator actions to control reactor water level are inherent to all events and not unique to the event in question. Because the EOP guidance is symptomatic in nature, the operator is simply instructed to monitor reactor vessel level and take actions to control it between an upper and lower limit. The action taken will consist of increasing injection or decreasing injection depending on the value and trend of the level indication. There is no procedural requirement to evaluate plant conditions to determine if a particular event may be in progress. As such, the operator actions to control vessel level are the same regardless of the event. Multiple redundant level indications are available to the operator as well as multiple mechanisms to add water.

The above statements follow from the basis for EOPs, which is, to the extent possible, to make them independent of a specific event.

Based on the foregoing, the NRC staff concludes that no new procedures are needed.

3.8.4 Training

With regard to training, Reference 7.2 states:

> Current expectations are that a discussion of the change and the bases for the change will be provided as part of the ongoing training program for operators. Included with this training, the new entry conditions for the Containment Control EOP would be discussed. Currently, it is not anticipated that any additional simulator training will be necessary.

The NRC staff concludes that the licensee has adequately addressed the issue of the Training with regard to incorporating this change, the bases for this change, the new entry conditions for the Containment Control EOP, and the simulator training into the training curriculum.

For reasons discussed above, the NRC staff concludes that the operators would properly implement the procedures and would take appropriate actions for the situation with the reactor cavity portion of the UCP drained.

Therefore, with respect to human performance, the TS changes proposed by the licensee are acceptable.

3.9 Evaluation of Proposed Changes to Technical Specification Surveillance Requirement 3.6.2.4.4

SR 3.6.2.4.4 of TS requires verification that all UCP gates are in the stored position or are otherwise removed from the UCP. This SR is applicable in MODES 1, 2, and 3.

The licensee has proposed revising SR 3.6.2.4.4 by adding the following NOTE:

> The requirements of this SR are not required to be met when all upper containment pool levels are maintained per SR 3.6.2.4.1 [upper containment pool level ≥ 23 ft.-3 in.] and suppression pool water level is maintained ≥18 ft.-5 1/12 in. (one inch above LCO 3.6.2.2 Low Water Level) [of 18 ft.-4-1/12 in. specified in LCO 3.6.2.2].

LCO 3.6.2.2 also specifies that the maximum suppression pool water level must be ≤18 ft.-9-3/4 in. This upper limit is not affected by the proposed revision.

The NOTE allows installation of the gates in Modes 1, 2, and 3. Draining the reactor cavity portion of the pool (see Section 3.1.2 of this SE) would only be permitted in MODE 3. The staff questioned the need to install gates prior to MODE 3. Reference 7.2 states that this allows any potential problems that may be encountered during gate installation to be resolved. Elsewhere in Reference 7.2, the licensee states:

> To realize the benefits...gate installation need only be performed just prior to or following initiation of plant shutdown for refueling or during planned maintenance activities. The time that the plant will be in MODE 3 operations with the reactor cavity drained is also limited and depends on planned outage activities.

There are four UCP gates covered by this NOTE: Gate 1 between the fuel storage pool and the fuel transfer pool, Gate 2 between the fuel storage pool and the reactor cavity pool, and the UCP weir wall Gates 4A and 4B between the separator pool and the reactor cavity pool. The staff questioned whether the NOTE was adequately specific, since it does not specify which gates could be installed. Reference 7.2 states that the safety analysis for the proposed gate installation described in Section 4.1 of the submittal supports installation of either Gate 1 or Gate 2. The analysis specifically considers Gate 2. This is the bounding case since the installation of Gate 2 results in the largest reduction in available suppression pool makeup system water irrespective of the installation of the other upper containment pool gates.

In order to compensate for the reduction in UCP water level, the licensee proposes to reduce the entrapped volume which is currently part of the Grand Gulf licensing basis.

Reference 7.1 proposes to take credit for the volume of equipment inside and below the top of the drywell weir wall (the drywell pool volume). The current TS water level limits do not include the volume of this equipment. The water, which collects inside the drywell below the top of the drywell weir wall, is not available for return to the suppression pool until the water level reaches the top of the weir wall and overflows into the suppression pool. By including the volume of equipment, Reference 7.1 proposes to reduce the amount of unavailable (entrapped) water that must be considered.

Reference 7.2 proposes to take credit for the following equipment:

- Concrete equipment foundations
- Galvanized grating
- Structural steel
- Steel pipe whip restraints
- Water filled piping (including recirculation system piping)
- HVAC [Heating, Ventilation, and Air Conditioning] duct work (only sheet metal thickness considered)

Reference 7.2 states that the total volume of this equipment is 863 ft^3. This is only 1.5 percent of the revised entrapped volume. However, the licensee points out that 863 ft^3 is equal to about 1.4 in. of suppression pool level. Therefore, if credit is not taken for the equipment volume, all other quantities of water (available and entrapped) remain as assumed by the licensee, and the initial suppression pool level is the minimum level allowed by the TSs, then the minimum suppression pool level would be 1.4 in. below the required post-LOCA water level of at least two feet above the top of the upper horizontal vent.

The current operating band between the low water level (LWL) and the HWL limits is 5-2/3 in. Since the low suppression pool level limit must be increased by one inch, the operating band is reduced to 4-2/3 in. This operating band could have been reduced further rather than taking credit for the equipment volume. This would result in an operating band of approximately 3.27 in. Rather than further reducing the operating band, the licensee takes credit for this equipment volume.

In response to a staff question about the assurance that the equipment volume would not be changed (decreased), the licensee states (Reference 7.2):

> The...credited equipment is fixed material identified on design basis drawings. Any changes to this equipment would involve a design change performed under approved Grand Gulf design change procedures in accordance with established design processes. These processes and procedures ensure that design changes receive review/concurrence from all affected design disciplines, including the Safety Analysis group responsible for this [suppression pool and UCP water inventory] analysis. Therefore, sufficient controls are in place to ensure that plant design changes that affect this equipment will receive Safety Analysis review for impact.

The staff finds the licensee's proposal of taking credit for equipment volume acceptable, and the proposed level of control over possible changes to the equipment volume is sufficient to ensure that changes to the facility as described in the FSAR are evaluated in accordance with regulatory requirements.

The licensee also proposes (Reference 7.1) a further reduction in the amount of water required to meet the design basis requirement of two feet above the top weir wall vent by revising the assumption that the reactor operator will fill the reactor vessel to the top of the vessel dome following a LOCA. The water filling the reactor vessel is not available to the suppression pool. Therefore, reducing the amount of water filling the reactor vessel reduces the entrapped water volume. Reference 7.1 proposes that the reactor vessel be assumed filled to Level 8. The Grand Gulf EOPs direct operators to maintain the vessel level at or below Level 8. Reference 7.2 states: "...operators are trained (classroom and simulator) to take prompt action to limit filling the reactor vessel above the Level 8 limit." The reactor vessel entrapment volume is then the volume required to fill the vessel from normal water level to Level 8. This represents another decrease in the entrapped (unavailable) water volume.

It is noted that, even though the operator is directed to maintain the reactor vessel level at Level 8, which is below the reactor vessel connection to the main steam lines, the licensee continues to assume that the main steam lines are flooded and that this water is not available to the suppression pool since "...the steam line piping is routed such that water that fills the steam lines would not drain back to the vessel after operators take action to reduce ECCS flow and decrease vessel level below the steam line elevation...." (Reference 7.2). Any water in the reactor vessel above Level 8 but not in the main steam lines is assumed to spill into the suppression pool, since the volume in the drywell pool is already considered full and is an entrapment volume.

Reference 7.2 discussed the operator's ability to control the reactor vessel water level at Level 8. The Grand Gulf EOPs direct the operator to maintain the level between Level 8 and the TAF which is at -167 in. Thus, the EOPs provide a broad control band in which to maintain reactor water level following a LOCA, which reflects possible operator difficulty in controlling the ECCS flow rate to obtain a fixed value of reactor vessel level. Since Level 8 is at the top of the operating band and the operating band is wide, the staff concludes that there is a reasonable assurance that Level 8 will not be exceeded.

Reference 7.1 does not change the allowance for containment spray hold-up on equipment and structural surfaces. This spray hold-up volume was part of the original Grand Gulf design basis. The staff concludes that the proposed changes do not impact this number.

The revised TS low and high suppression pool water level limits are nominal values. Reference 7.2 explains that the uncertainty associated with the narrow-range instruments used to measure these water levels is included in the surveillance procedures. Since the licensee will continue to use the same instruments to measure the suppression pool water level, the same instrument uncertainties, applied in the same way, will apply to the proposed suppression pool water level limits. The staff finds this acceptable.

The licensee also points out that there are alternate water sources which are not safety-related but which could be used, if available, to add water to the reactor vessel and, hence, to the suppression pool. These include feedwater injection and high pressure core spray injection from the condensate storage tank. In addition, there is a safety-related water source; the standby service water system.

Based on the discussion above, the staff finds the licensee’s proposal to install gates in the UCP in MODES 1, 2, and 3, and the associated proposed change to SR 3.6.2.4.4, acceptable.

3.10 Evaluation of Proposed Changes to Technical Specification 3.10.9

The purpose of the gate installation discussed in Section 3.9 herein is to permit draining of the reactor cavity portion of the UCP in MODE 3. Reference 7.1 proposes that this is acceptable when the reactor pressure is less than 230 psig and the reactor has been subcritical > 3 hours. It is noted that proposed LCO 3.10.9.c requires that the level in the fuel storage and transfer canal portions of the UCP must be maintained above 23 ft.-3 in.

Reference 7.1 compensates for draining the reactor cavity pool by a combination of reducing the assumed amount of entrapped water and increasing the minimum suppression pool level. In addition to these assumptions, the licensee has performed calculations with the GOTHIC computer code to demonstrate that the containment sprays will not automatically actuate for a large break LOCA in MODE 3, since the containment pressure with the reactor vessel at 230 psig and the reactor subcritical for > 3 hours will not result in the containment pressure reaching the containment spray setpoint. Therefore, the allowance for containment spray hold-up on equipment and structural surfaces is not considered in the calculation of entrapped volume.

Reference 7.1 reduces the entrapped water volume by including the volume of equipment in the drywell pool and assuming refill of the reactor vessel to Level 8 rather than to the vessel dome.

Even with the reduced entrapment volume, Reference 7.1 calculates that the water level in the suppression pool must be increased to a LWL limit of 20 ft.-0 in. and a HWL limit of 20 ft.-6 in., which provides an operating margin of 6 in. The limits also include a one inch measurement uncertainty. In addition to this one inch measurement uncertainty, Reference 7.1 increases the HWL limit to 20 ft.-7 in. for the supporting calculations.

In order to provide more operational flexibility, Reference 7.1 includes TS Figure 3.9.10-1 which shows the acceptable region in a plot of suppression pool level as a function of the UCP level. Reference 7.2 provides the calculations which are the basis for this plot. The staff has reviewed these calculations and finds them acceptable, as set forth below.

The licensee performed calculations to demonstrate that, for the reactor conditions of 230 psig and subcritical for 3 hours, containment spray will not actuate following a LOCA. This is to support the assumption that the entrapped water, due to hold-up of spray on equipment and structural surfaces, does not have to be considered for the case of a LOCA in MODE 3 with the reactor cavity portion of the UCP drained. These calculations were performed using the GOTHIC computer code. Reference 7.2 describes the Appendix B programs of both the licensee and Numerical Applications Inc., who developed the code and maintains the code for the Electric Power Research Institute. The following excerpt from Reference 7.2 describes the licensee's commitments under its Appendix B program for procured software:

> [Grand Gulf Appendix B procedures for procured software] require that the code installation and verification and validation be formally documented in a Computer Program Documentation Package. Verification and validation is accomplished by execution of sample problems and comparison of results to those provided by the code developer. Procedures also delineate qualification requirements for users and tracking of code error notices supplied by the code developer.

Error reporting is also required by 10 CFR Part 21, "Reporting of Defects and Noncompliance."

Attachment 4 to Reference 7.1 provides a description of the use of GOTHIC for this application and the results of GOTHIC predictions of containment conditions resulting from a MODE 3 MSLB.

The decay heat is calculated based on bounding ORIGEN 2.1 calculations with cycle-specific inputs. Reference 7.2 describes the conservatism in the decay heat calculation. The staff finds the licensee's prediction of decay heat to be conservative and acceptable.

The containment heat sinks are those given in Table 6.2-9 of Reference 7.4. The Uchida condensing heat transfer coefficient correlation is used to determine heat flow to structural heat sinks. This is consistent with usual practice in conservative licensing calculations.

The GOTHIC computer code was used not only to calculate the containment conditions following a LOCA, but also to predict the mass and energy flow of reactor coolant and ECCS water discharged from the postulated pipe break. Reference 7.2 provides more information on the use of GOTHIC computer code for break flow calculations. The licensee benchmarked the reactor vessel blowdown model to General Electric Company (GE) MSLB calculations for full power conditions, shown in Reference 7.4, Table 6.2-11 and Figure 6.2-19. This model incorporates the conservatism and assumptions used in the approved GE methods. The results of the benchmarking are given in Attachment 4 to Reference 7.1. The comparisons with peak pressure and temperature values show good agreement. Differences between the calculations in the rate of increase and decrease of the pressure and temperatures are satisfactorily explained in Reference 7.2. In particular, the agreement between Reference 7.4 and GOTHIC computer code predictions of peak containment pressure is good. This is a good indication of agreement between GOTHIC and Reference 7.4 reactor vessel blowdown calculations.

An important consideration in BWR peak pressure calculations is the modeling of flow through the vents. Reference 7.2 states that

> The GOTHIC vent flow models include all phenomena captured in the GE model, including fluid inertia, irreversible loss factors, vent area effects on flow, and choked flow based on a homogeneous equilibrium model. The only significant difference between the GOTHIC and GE models is the vent clearing model. The GE method does not allow any water to flow through a vent until the level in the weir annulus is depressed to the center line of the vent. GOTHIC cannot reproduce this clearly non-physical approach. However, benchmark sensitivity calculations show that this effect does not significantly influence the vent clearing times or drywell pressurization rates.

Reference 7.2 listed several conservative assumptions in the calculation. The safety analyses were done using a reactor pressure of 235 psig. A reactor pressure of 230 psig is the allowable value in the TSs. The higher pressure results in a higher mass flow rate from the vessel. Reference 7.2 states that this 235 psig pressure was selected based on the results of all the analyses supporting these TS changes and is not derived from any one of them.

Reference 7.1 assumed that the reactor cooled down and depressurized to 235 psig by use of the S/RVs rather than the main condenser. This results in a higher suppression pool temperature and hence, a higher containment pressure.

The suppression pool water level is assumed to be at the LWL of the TSs. This would usually be conservative since the more typical situation would be an operational suppression pool level greater than the TS LWL limit.

The initial temperature of the suppression pool is assumed to be 110 °F, even though the TSs require this temperature to be less than or equal to 95 °F.

In addition, Reference 7.1 assumes that the MODE 3 initial condition of 235 psig is reached with the reactor coolant saturated (i.e., no credit is taken for the normally subcooled liquid that would be supplied to the vessel during the cooldown). The saturation temperature corresponding to a pressure of 235 psig is 401 °F. Saturation conditions are bounding for determining the amount of water which must be added to the vessel to maintain Level 8 because this provides the lowest water density, which in turn produces the largest possible level shrink during the depressurization from 235 psig to 0.0 psig.

The licensee was requested to consider the possibility that the operator might manually initiate the containment spray following a LOCA based on indications of increasing containment pressure and temperature, even though calculations show that the spray setpoint pressure will not be reached. This would result in water entrapment not accounted for in the design basis. Reference 7.2 states that manual initiation is not prohibited, but described the very specific criteria in the EOPs for initiating spray. The GOTHIC computer code calculations show that (1) none of the criteria for manual initiation of containment sprays specified in the EOPs is reached, and (2) the increases in containment pressure and temperature are slow enough to provide assurance to the operator that the criteria for manual containment spray initiation will not be reached.

Based on the above, the staff finds the application of the GOTHIC computer code and the results obtained to be conservative and acceptable. Therefore, the spray hold-up on equipment and structural surfaces need not be included in determining entrapped volumes since this conclusion is based on conservative assumptions and calculations.

Furthermore, based on the staff's review of the licensee's calculations and conservative assumptions, as set forth above, the staff finds proposed TS 3.10 "Special Operations," TS 3.10.9 "Suppression Pool Makeup-MODE 3," acceptable.

4.0 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.

5.0 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 (April 30, 2002 , 67 FR 21289), and there has been no public comment on such finding. 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.

6.0 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.

7.0 REFERENCES

7.1 GNRO-2002/00011, William A. Eaton (Entergy) letter to NRC, "Grand Gulf Nuclear Station, Unit 1, License Amendment Request New Special Operations LIMITING CONDITION FOR OPERATION Suppression Pool Makeup-MODE 3 (LDC 2002-006)," dated February 25, 2002.

7.2 GNRO-2002/00072, Jerry C. Roberts (Entergy) letter to NRC, "Grand Gulf Nuclear Station, Unit 1, Supplement to Amendment Request, Response to Request for Additional Information Concerning New Special Operations LIMITING CONDITION FOR OPERATION Suppression Pool Makeup-MODE 3 (LDC 2002-006)," dated August 16, 2002.

7.3 GNRO-2002/00077, Jerry C. Roberts (Entergy) letter to NRC, "Grand Gulf Nuclear Station, Unit 1, Supplement 2 to Amendment Request, Response to Request for Additional Information Concerning New Special Operations Limiting Condition for Operation Suppression Pool Makeup-MODE 3 (LDC 2002-006)," dated August 22, 2002.

7.4 Grand Gulf Nuclear Station, Unit 1, Updated Final Safety Analysis Report (UFSAR).

7.5 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," 1975, Section 6.2.1.1.c.

7.6 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

7.7 NUREG-0978, "Mark-III LOCA Related Hydrodynamic Load Definition," August 1984.

7.8 Letter dated May 8, 1982, from Mr. John Humphrey, a former General Electric Company engineer, to the Grand Gulf Nuclear Station licensee, regarding safety concerns about the Mark III containment.

7.9 Supplements 2, 3, and 4 to NUREG-0831, "Grand Gulf Nuclear Station Safety Evaluation Report."

7.10 Letter dated March 23, 1987, from Lester L. Kintner, USNRC, to Oliver D. Kingsley, Jr., System Energy Resources, Inc., regarding NRC's evaluation.

7.11 Entergy Letter to USNRC dated October 22, 1993.

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Date: September 6, 2002

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