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GNRO-2002/00072

August 16, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
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)

Dear Sir or Madam:

By letter GNRO-2002/00011 dated February 25, 2002, Entergy Operations, Inc. (Entergy) proposed a change to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TS) to add a new Special Operations LIMITING CONDITION FOR OPERATION (Suppression Pool Makeup-MODE 3) to allow installing Upper Containment Pool (UCP) gates and draining the reactor cavity pool portions of the UCP while still in MODE 3, "Hot Shutdown," with the reactor pressure less than 230 pounds per square inch gauge (psig). The request would also modify the applicability of the UCP gates surveillance requirement to allow installation of UCP gates in MODE 1, "Power Operation," MODE 2 "Startup," or MODE 3. The proposed change would allow early gate installation and allow draining of the pool while holding the plant in MODE 3 to facilitate starting of certain outage functions.

Entergy and members of your staff held a call to discuss specific questions regarding the performance of these tests. As a result of the call, 41 questions were determined to need formal response. Entergy's response is contained in Attachments 1 and 2. Attachment 3 is the revised proposed mark-up for the Bases of SR 3.6.2.4.4.

There are no technical changes proposed. The original no significant hazards considerations included in Reference 1 is not affected by any information contained in the supplemental letter. There are no new commitments contained in this letter.

If you have any questions or require additional information, please contact Bill Brice at 601-368-5076.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed August 16, 2002

Sincerely,

JCR/WBB/amt

Attachments:
1. Response to Request For Additional Information
2. Sample Calculations
3. Revised Markup of Technical Specification Bases Pages

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Attachment 1

To

GNRO-2002/00072

Response to Request for Additional Information

Response to Request for Additional Information
GRAND GULF TAC MB4260

1. **Proposed Technical Specification (TS) 3.10.9 allows draining the "upper containment pool." Actually, isn't it intended that only a portion of the upper containment pool (UCP) is to be drained (the separator pool partially and the reactor cavity pool completely)?**

   a. **Should the TS be more specific?**

   TS 3.10.9.c address draining only a portion of the UCP. The term "Upper Containment Pool" in the proposed TS Limiting Condition for Operation (LCO) 3.10.9.b and Figure 3.10.9-1 is used generically to refer to the portion of the UCP that can be drained. TS LCO 3.10.9.c specifically requires that the level in the fuel storage pool and transfer canal areas of the UCPs be maintained to a minimum 23 ft 3 inches, which is the current TS limit for UCP level The specification is adequate as written.

   b. **How is the draining done?**

   The draining can be done in any way so long as the UCP and suppression pool levels are within the limits defined by TS Figure 3.10.9-1. One purpose of including this figure was to avoid specifying exactly how the UCP drain down must proceed. So long as the levels are in the acceptable range of this figure, the plant configuration is supported by the safety analyses performed for the MODE 3 drain down. Currently the draining is controlled by an existing Operations procedure. The procedure uses existing piping and valves. Temporary installed piping is not used. Most of the water from the UCP will be drained to the suppression pool. Any water that would exceed the Figure 3.10.9-1 limits for suppression pool level would be drained elsewhere (e.g., the Refueling Water Storage Tank).

   c. **Page 2/19 of the February 25, 2002 submittal, Section 3.0 states that to optimize scheduling, Grand Gulf Nuclear Station (GGNS) desires to install UCP "gates." Isn't Gate 2 the only gate to be installed? If only gate 2 is to be installed in MODES 1, 2, or 3, why isn't the proposed note accompanying surveillance requirement (SR) 3.6.2.4.4 more specific? Additional detail in the surveillance requirement is not necessary if the Bases or Updated Final Safety Analysis Report (UFSAR) adequately specify which gates will be installed.**

   Note: Refer to Figure 1, page 15/19, Attachment 1of the February 25, 2002 submittal for the location and identification of the UCP gates discussed in the response below.

   There are four UCP gates referenced by TS SR 3.6.2.4.4: Gate 1 between the Fuel Storage Pool (FSP) and Fuel Transfer Pool (FTP), Gate 2 between the FSP and the reactor cavity pool, and UCP weir wall Gates 4A and 4B. The safety analysis for the proposed gate installation described in Section 4.3.1 of the submittal supports installation of either Gate 1 or 2. The analysis specifically considers Gate 2. Installation of Gate 2 is bounding relative to installation of Gate 1 due to the larger reduction in Suppression Pool Makeup (SPMU) System volume. The 1 inch increase in the suppression pool minimum level proposed for the gate installation evolution offsets (together with the other considerations described in Section 4.3.1) the SPMU volume loss due to installation of Gate 2. Installation of Gate 1 isolates a smaller volume of the UCP and thus results in smaller reduction in SPMU System makeup volume. Therefore, since Gate 2 installation would have the same effect as Gate 1 installation on that portion of the UCP, the proposed change to TS SR 3.6.2.4.4 does not specify which of these gates (1 or 2) can be installed.

The UCP weir wall gates (4A and 4B) are below the normally required level of the UCP water level. The proposed note accompanying SR 3.6.2.4.4 explicitly requires that the UCP levels be maintained per SR 3.6.2.4.1. SR 3.6.2.4.1 requires the UCP water level to be ≥ 23 feet 3 inches above the pool bottom. This elevation is above the top of the weir wall gates. Therefore, Installation of gates 4A and 4B after the UCP has been flooded was not considered practical under the proposed TS note. The attached proposed TS Bases change is intended to clarify the TS and specifically identifies the gates that can be installed and explicitly excludes the weir wall extension gates.

2. **Explain why it is necessary to install the gates in MODES 1 or 2 rather than in MODE 3 prior to draining the reactor cavity pool. What would be the time savings over the present TS requirements if Gate 2 were added while in MODE 3 as compared to installing the gates in MODES 1 or 2?**

   The advantage to performing this activity while still in MODES 1 or 2 is that it allows allocation of resources to be focused on tasks that are *required* to be performed during outages, due to conditions or requirements. Additionally, this allows time to resolve any potential problems that may be encountered during installation of the gates. The estimated time from the start of MODE 3 until reactor pressure is below the specified 230 pounds per square inch gauge (psig) is approximately 4 hours. The analysis supports installing the gates in MODES 1, 2 or 3.

3. **The proposed TS allows the gates to be installed in MODE 1 or MODE 2 at any time during an operating cycle, or for the entire cycle. Why shouldn't a time-before-shutdown limit or a total time during an operating cycle limit be proposed for operation in MODES 1 and 2 in this configuration?**

   The analysis performed to support this change determined that as long as the suppression pool level is maintained at > 18'-5 1/12", then adequate makeup volume is available to support installation of the gates in MODES 1 and 2. Outage preparation and planning is very complex and requires considerable attention to detail. Any work that can safely be done outside of the outage assists in managing the workload and preventing unforeseen problems and delays. As stated above, the analysis supports performing this work in MODES 1, 2, or 3 as long as SRs are met. Therefore, this work can be performed safely at any time as long as all associated requirements are in place.

4. **Verify that the Required Action for LCO 3.10.9 Condition B requires removing Gate 2 and restoring level to the upper containment pool level required by SR 3.6.2.4.1.**

   The required action for Condition B states that the "suspended MODE 3 requirements" must be complied with. This requires that the upper pool levels be restored within the 12 hours allowed by the LCO and requires restoring the suppression pool levels to their normal levels. If this can be accomplished without removing Gate 2, then this would be acceptable as proposed SR 3.6.2.2.4 allows the gate to be installed with suppression pool level > 18' 5 1/12".

5. **The operator will control reactor vessel level to Level 8 following a Loss of Cooling Accident (LOCA).**

   a. **What is the band on operator action for Level 8? Why is this band acceptable?**

   The initial Emergency Operating Procedures (EOP) guidance directs that the Reactor Pressure Vessel (RPV) level be controlled between Level 8 (+53.5 in) and Level 3 (+11.4 in). The analysis assumes RPV injection is limited to Emergency Core Cooling System (ECCS) pumps. Since ECCS is designed to supply large volumes of water to the RPV, the throttling capabilities are limited to opening and closing the injection valve. Due to high ECCS flow rates, only gross level control in the RPV can be achieved. RPV level fluctuations would likely exceed the directed control band.

   The EOP then asks if RPV level can be maintained within the initial band. Under the assumed conditions the operator would answer "No" and move to the next level of guidance in the EOP. The next EOP guidance expands the RPV level control band to between Level 8 (53.5 in) and Top of Active Fuel (TAF) (-167 in). This expanded control band now allows the operator to establish a reasonable control band within the ECCS control capability without undue challenge to the ECCS system. Although the allowable control band is expanded to TAF to address the broadest spectrum of events, the operators are trained to only increase the control band to the point necessary to minimize the challenge to ECCS. Typically the operator will establish a control band between Level 8 (53.5 in) and Level 2 (-41.6 in) which is well above TAF.

   Following initial RPV fill, ECCS injection valves would be closed to control level. With the ECCS injection valves closed, RPV level would decrease due to steaming losses from decay heat. It would require approximately 30 minutes for RPV level to decrease from Level 8 to Level 2. Refilling the RPV to Level 8 would only require one ECCS injection valve to be opened for less than 5 minutes. This cycle would be repeated as required to maintain RPV level.

   These control bands are acceptable because the Grand Gulf EOPs are developed from the Boiling Water Reactor Owners Group (BWROG) Emergency Procedure Guidelines (EPGs). The BWROG EPGs were reviewed and approved for use by BWR licensees through a Safety Evaluation Report (SER) dated 9/12/1988.

   b. **What considerations went into making the assumption for sizing the upper containment pool that the vessel would fill to the top of the dome? Why is it now acceptable to neglect these considerations?**

   The original sizing of the UCP, assumed here to refer to the portion of the UCP available to the SPMU System, accounts for "all conceivable post-accident entrapment volumes (i.e., places where water can be stored while maintaining long-term drywell vent water coverage)" (GGNS UFSAR Section 6.2.7). One entrapment volume considered is the 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 (UFSAR Section 6.2.7.1.g). By considering all "conceivable" entrapment volumes, a very restrictive and bounding UCP size can be established with no consideration of actual post-accident plant response or EOP directed operator actions.

The safety analysis supporting the proposed gate installation and MODE 3 reactor cavity drain evolutions (UFSAR Section 4.3.1, Attachment 1 of the February 25, 2002 submittal) does not consider the water needed to fill the reactor vessel to the top of the dome. The reactor vessel entrapment volume considers only the volume required to fill the vessel from normal water level (36.7 inches) to Level 8 (53.5 inches). The volume required to fill the vessel above Level 8 is neglected based on considerations of realistic plant accident response and proceduralized operator actions. These considerations include:

- Break size and location.
- The existence of EOPs directing operators to control post-accident reactor vessel level at or below Level 8.
- The likelihood that the available ECCS systems will be capable of flooding the vessel to the top of the dome following a Design Basis Accident (DBA) LOCA. This capability depends on the number of ECCS pumps available, the size and location of the break, and the reactor vessel pressure.

Post-accident plant response is documented in the containment DBA LOCA analyses in UFSAR Section 6.2 and by the MODE 3 containment accident analyses performed for the proposed MODE 3 reactor cavity drain evolution. Operator actions are proceduralized in the GGNS EOPs. The analyses supporting the proposed TS changes consider operator action to control reactor level at or below Level 8.

The assumptions that the vessel is flooded to Level 8 rather than the top of the dome is applicable only for the proposed gate installation and MODE 3 reactor cavity drain evolutions. The time that the plant will be in one of these two configurations is limited. To realize the benefits cited in Section 3.0, Attachment 1 of the submittal, 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 (e.g., noble metal addition).

c. What is the volume of the flooded steam lines?

The volume of the flooded steam lines is 1,136.44 ft^3. This volume includes the volume in the steam lines out to the first Main Steam Isolation Valve (MSIV) for three lines and out to the second MSIV on one line. This is consistent with the original design basis as described in UFSAR Section 6.2.7.1.g.

d. Page 6/19 states that the volume of the steam lines must be taken into account because the steam lines may fill prior to the operators taking action to reduce ECCS flow. The assumption appears to be made that the ECCS water injected into the vessel above Level 8 subsequently leaves the vessel through the break and returns to the suppression pool; that is, the break water remains available and is not transported to an inaccessible volume (e.g., the drywell pool). Won't at least some of this water flow to the drywell pool and not remain available? If this is so, shouldn't this volume be considered in addition to the volume in the steam lines?

The assumption made in calculating the containment and Suppression Pool Makeup System water requirements (Section 4.3.1, Attachment 1 of the February 25, 2002 submittal) is that any ECCS water injected into the vessel that increases the level above Level 8 subsequently leaves the vessel through the break and returns to the drywell pool. The drywell pool is postulated to fill to the top of the weir wall and overflow into the suppression pool. At this point, any additional ECCS water spillage from the break returns to the suppression pool. Thus, the water required to fill the drywell pool to the top of the weir wall is considered an entrapment volume in the analyses supporting the proposed changes. After the drywell pool forms, any additional water spillage from the break is water that will return to the suppression pool.

The volume of water required to fill the steam lines is also considered a water entrapment volume in the analyses supporting the proposed changes. The volume includes the water required to fill the steam lines out to the inboard MSIV on three lines and out to the outboard MSIV on the remaining line. This volume is considered because the steam line piping is routed such that water that fills the steam lines would not drain back into the vessel after operators take action to reduce ECCS flow and decrease vessel level below the steam line elevation.

6. **In the reactor cavity drained (MODE 3) scenario, a GOTHIC calculation shows that the containment spray set point is not reached.**

   a. **What prevents the operator from manually initiating sprays before the automatic setpoint if he/she sees a continuing increase in containment pressure?**

   The referenced scenario is a bounding analysis assuming long-term suppression pool at the minimum allowable level to maintain 2 feet of vent submergence (14.5 feet). This assumption maximizes the long-term suppression pool temperature and containment pressure. The analysis shows that no spray initiation, either manual or automatic, is required to protect the design pressure (15 psig). The containment spray setpoint is not reached, eliminating concerns of an automatic spray initiation. However, containment sprays are available and manual spray initiation is not prohibited.

   Manual initiation of containment sprays is controlled through the EOP guidance. The EOP guidance regarding manual initiation of containment sprays is based on the response to two questions. The first question is: "Can containment temperature be maintained below 185°F?" If answered "No" EOP guidance would then direct manual initiation of containment sprays.

   In the referenced scenario a peak containment temperature of 140.1°F is calculated to occur approximately 8 hours into the event. Thus the calculated peak containment temperature is approximately 45°F below the EOP limit. Combined with the slow increase in containment temperature, the operator could easily determine that containment temperature would remain below 185°F.

   The second question is: "Can containment pressure be maintained in the Safe Zone of the PSP (Pressure Suppression Pressure)?" If answered "NO" EOP guidance would then direct manual initiation of containment sprays.

   In the referenced scenario a calculated peak containment pressure of 5.94 psig is expected to occur approximately 9.4 hours into the event. The EOPs also require suppression pool level to be maintained within the proposed MODE 3 TS limits. With suppression pool level between 20 feet and 20 feet 6 inches as required by the proposed suppression pool level LCO, the PSP limit is approximately 8.5 psig. Thus the calculated peak containment pressure is approximately 2.5 psig below the EOP limit. Combined with the slow increase in containment pressure the operator would determine that containment pressure would remain below 8.5 psig.

Thus in the referenced scenario, when EOP actions are considered, the response to both manual spray initiation questions would be "Yes" and the EOP guidance would not direct manual initiation of containment sprays.

b. How long after such a manual spray initiation would the operator determine that containment sprays are unnecessary and terminate the spray (including operator action, valve closures, etc.)?

As stated in the response to Item 6 (a), EOP guidance would not direct the operator to manually initiate containment sprays. However, to answer the question in a general sense, securing of containment sprays is based on the response to the two questions discussed in the response to Item 6 (a). The operator continues to evaluate the two questions after containment sprays are initiated. When the response to both questions becomes 'Yes', the EOPs would no longer direct manual operation of containment sprays and they would be secured. In addition, EOP guidance directs that containment sprays be terminated before containment pressure decreases to 0 psig. Because these are judgment calls it is impossible to specify a fixed time that containment sprays would remain in operation.

c. What volume of water would be lost during this time? Would adequate water remain in the suppression pool?

As stated in response to items 6(a) and 6(b), EOP guidance would not direct the operator to manually initiate containment sprays and the time containment sprays would be operated if manually started would vary depending on plant conditions. The volume of suppression pool water entrapped in the drained reactor cavity following spray operation and the resulting suppression pool level depend on the time of spray initiation, the number of spray trains started and the duration of spray operation.

7. The proposed TS changes would necessitate adding water to the suppression pool at some time after the LOCA.

a. Do current Grand Gulf design basis long term containment analyses require water addition to the suppression pool?

The current Grand Gulf design basis long-term containment (LOCA) analyses documented in UFSAR Chapter 6.2 do not require water addition to the suppression pool. The safety analyses supporting the proposed TS changes also do not require adding water to the suppression pool at any time after a DBA LOCA (i.e., Main Steam Line Break, Recirculation Line Break) or following the Small Break with Bypass Leakage capability analysis. The larger (2.5 ft^2) steam line break with bypass leakage analysis considered in response to Humphrey Issues 5.1 and 9.2 does not require water addition to the suppression pool if operators reduce ECCS flow and prevent liquid spillage from the break. This is the assumption in the analysis of this event discussed in UFSAR Section 6.2.1.1.5.5. Water addition to the containment from external sources is required when certain EOP actions are considered (see response to question 6(a) above). This is true for both the current DBA LOCA analyses and for the analyses supporting the proposed gate installation and MODE 3 reactor cavity drain down evolutions.

The events analyzed for the proposed changes that identify a requirement for post-LOCA makeup to the suppression pool are large breaks (2.5 ft^2 Humphrey's break and 3.54 ft^2 Main Steam Line Break (MSLB)) with bypass leakage assuming delayed operator response such that some ECCS liquid spills from the break. The bypass leakage is equal to the Grand Gulf licensed effective area ($A/\sqrt{k}$) of 0.9 ft^2. Bypass leakage analyses are special calculations that evaluate the bypass leakage capability of the containment considering containment sprays and containment heat sinks (see discussion in UFSAR Section 6.2.1.1.5.5). The concept of the Grand Gulf Mark III pressure suppression containment is that any steam released from the primary system will be condensed by the suppression pool and will not have an opportunity to produce a significant pressurization effect on the containment. Bypass leakage refers to a leakage path between the drywell and containment. The leakage path bypasses the suppression pool allowing the leaking steam to pressurize the containment. The bypass leakage area is expressed as the effective, total leakage area (A) divided by the square root of the total irreversible loss coefficient of the leakage flow path (k). This parameter ($A/\sqrt{k}$) characterizes the size of the leakage path and is derived from steady, incompressible fluid flow theory. The Grand Gulf licensed bypass leakage area is 0.9 ft^2 and is based on the results of a small (0.07 ft^2) steam line break with bypass leakage crediting one containment spray loop (at 13 minutes) and containment heat sinks. This analysis is described in Section 6.2.1.1.5.5 of the UFSAR. In response to Humphrey Issues 5.1 and 9.2, a sensitivity study of various break sizes from a small break to a DBA was conducted to determine containment pressurization prior to initiation of containment sprays. The most limiting break was determined to be a 2.5 ft^2 steam line break. Results from this analysis show containment pressurization consistent with the small break size discussed in the UFSAR.

The small break with bypass leakage analysis establishes the licensed bypass leakage area and is therefore considered the design basis bypass leakage analysis. The 2.5 ft^2 steam line break is a sensitivity analysis that supports the design basis small break results assuming steam flow from the break. However, due to possible delayed operator response times, the larger breaks with bypass leakage are the most limiting analyses for the proposed MODE 3 reactor cavity drain evolution in terms of suppression pool entrapment volumes. The concern is that operators will not reduce ECCS flow to the reactor vessel before some liquid spills from the break. However, steam leaking from the drywell directly to the containment during the initial vessel blowdown could cause containment pressurization and actuation of containment sprays. Given the large spray hold up volume created by the drained cavity, spray actuation in MODE 3 with the reactor cavity drained and water entrapped in the drywell pool could result in a requirement for post-LOCA makeup water to the suppression pool to ensure a minimum long-term suppression pool vent submergence of 2 feet

To address the spray entrapment concern, the 3.54 ft^2 MSLB with 0.9 ft^2 bypass leakage event described in Section 4.3.7 of Attachment 1 of the submittal was considered. This event was developed to bound the long-term suppression pool inventory requirements for a LOCA in MODE 3 with the reactor cavity drained. As such, it is a special, beyond design basis event designed to create the most limiting long term containment water inventory requirements and identify response times available to the operators for securing external makeup to the suppression pool. The event assumes a guillotine break of the main steam line inside the drywell, the maximum licensed suppression pool bypass leakage path, maximum ECCS pumps, and delayed operator response to control level to allow some liquid spillage from the break and partial drywell pool formation. The break size and ECCS pump assumptions maximize the rate of inventory loss from the break (rate of entrapment in the drywell pool) and the spray flow rate (rate of entrapment in the drained cavity). The MSLB with bypass leakage events characterize the available operator response times for providing external makeup to the suppression pool.

b. **What other actions would the operators be doing during the time that they will have to refill the suppression pool?**

Operators will be taking actions to maintain Reactor level to within modified bands, in accordance with the EOPs as described in the response to Question 5. Operators would also be taking actions to ensure containment integrity is maintained post-LOCA in accordance with the EOPs. Suppression pool level control is a part of containment control.

8. **Is Item I. of the Suppression Pool Makeup System Design Bases (Section 6.2.7 of the Grand Gulf UFSAR) equivalent to the Dump Time criterion discussed in Section 4.3.8 of the February 25, 2002 submittal? If not, please demonstrate that Item I. is satisfied.**

Yes.

9. **Please explain how item f. of UFSAR Section 6.2.7.1, Design Basis, is satisfied. It appears that it is not satisfied for those events described in Section 4.3.7 of the February 25, 2002 letter since suppression pool makeup is necessary.**

Item f of UFSAR Section 6.2.7.1 refers to containment water inventory and distribution requirements. Item f is satisfied by the safety analyses performed for the proposed TS changes described in Attachment 1, Section 4.3.1 of the submittal. The proposed suppression pool level requirements established by this analysis ensure that, for any design basis accident, the suppression pool volume between the proposed low water level limits and the minimum post-accident level, plus the makeup volume available from the upper pool is adequate to supply the post-accident entrapment volumes. No external suppression pool makeup is necessary for any of the MODE 3 DBA analyses. External suppression pool makeup is only required for the larger break LOCAs with licensed bypass leakage. The most limiting of these events is the special, beyond design basis Main Steam Line Break with Bypass Leakage analysis described in Section 4.3.7 of the submittal. As discussed in response to Question 7a, this is not a DBA but is part of a sensitivity study to demonstrate the times required for manual operator actions are reasonable.

10.

a. **Has a beyond-design-basis analysis (perhaps realistic) been done of the consequences of completely losing the UCP inventory following a LOCA?**

No. The Grand Gulf Mark III containment design relies upon a portion of the water stored in the upper containment pools. This water is redistributed by gravity to the suppression pool by the Suppression Pool Makeup (SPMU) System under post-accident conditions. See the discussion in Attachment 1, Section 4.3 (page 4/19) of the February 25, 2002 submittal.

b. If so, what are the results?

This question in not applicable as discussed above.

c. What design basis limits would be exceeded?

Without the UCP inventory, the minimum post-accident suppression pool level requirement would not be met. The containment design basis requires that a minimum post-accident suppression pool water level of at least 2 feet above the top row of the LOCA vents be maintained. This minimum level ensures that the suppression pool contains adequate post-accident inventory to perform the pressure suppression function. For a large break LOCA without makeup from the UCPs, the suppression pool level would quickly decrease below this level and the 2 foot submergence criteria would not be met during a portion of the reactor vessel blowdown when pressure suppression is required. A lower long-term suppression pool volume could also result in inadequate Net Positive Suction Head (NPSH) for the ECCS pumps. The additional water from the UCP is also credited as part of the long term suppression pool heat sink. Sizing of the residual heat removal heat exchanger takes credit for the additional UCP water mass in the calculation of the post-LOCA peak containment pressure and suppression pool temperature.

11. What is stored in the upper containment pools in MODES 1, 2 and 3?

Blade guides, new blades (possible), refueling tools, and other non-irradiated components. There is no prohibition on storing irradiated components such as old blades or LPRM dry tubes, but storing of spent fuel is prohibited.

12. What other activities will be occurring in and around the upper containment pool in MODE 3 at the same time that the upper containment pool is being drained?

The following general activities are planned during the time that the cavity level is less than 23 feet:

- Refueling Cavity decontamination
- Equipment surveillances and testing (i.e., Refueling Bridge)
- Radiation Protection surveys and sampling
- Qualification of personnel on refueling equipment

13.

a. **Please list any conservative assumptions made in this analyses or other conservative factors that provide some defense-in-depth (including the GOTHIC analyses).**

Conservative assumptions and conservative factors used in the analyses of the required SPMU System (UCP) volumes include those used in the original development of the SPMU design basis. These include:

- Consideration of all realistic water entrapment volumes, including water entrapped in all four main steam lines and the water required to fill the drywell pool.

- No consideration for alternate makeup sources such as feedwater injection or HPCS/RCIC injection from the Condensate Storage Tank.

- Maintaining the SPMU System operable during both the proposed gate installation and MODE 3 reactor cavity drain down evolutions.

- Post-accident suppression pool level based on initial pool levels at the lower analytical limits considered in the analyses (actual pool levels are normally maintained above these minimum analytical values).

The GOTHIC analyses supporting the proposed MODE 3 reactor cavity drain evolution use containment models benchmarked to the current licensed containment analyses performed by GE and documented in Section 6.2 of the UFSAR. As such, these models incorporate the conservatisms included in the approved GE methods. Deviations from input assumptions used in the GE analyses (e.g., decay heat models) result in more realistic but bounding models. These deviations are documented in Attachment 4 of the February 25, 2002 submittal. Conservatisms in the GOTHIC analyses also include:

- Initial suppression pool volume that neglects the volume of water in the drywell vents.

- Initial suppression pool temperature equal to 110°F (the proposed TS requires pool temperature to be ≤ 95°F.

- Initial Upper Containment Pool temperature equal to 125°F (current TS SR 3.6.2.4.2 limit).

- Conservative MODE 3 reactor vessel initial conditions. The initial conditions are calculated assuming a controlled vessel cooldown to the 235 psig analytical vessel pressure limit with liquid in the vessel at saturated conditions. This assumption does not credit the normally cooler (subcooled) liquid that would be supplied to the vessel during the cooldown evolution. The cooldown rate is assumed to be 50°F/hr.

- Post-accident reactor cooldown via the Safety/Relief Valves (SRVs) to the suppression pool is assumed in the MODE 3 bypass leakage calculations. The current bypass leakage analysis assumes cooldown is performed using the main condenser. This assumption increases the temperature of the suppression pool, which increases the containment pressure.

Other conservatisms include:

- A 5 psi margin in the proposed TS allowable reactor vessel pressure. The safety analyses support a MODE 3 reactor pressure vessel equal to 235 psig. The proposed TS requires reactor pressure to be < 230 psig.

- A high containment pressure spray setpoint permissive based on the lowest spray setpoint. This setpoint is equal to the nominal value less the total uncertainty in the setpoint loop. The calculated MODE 3 MSLB DBA peak pressure is 1.6 psig below this lower analytical setpoint, providing additional margin to an automatic spray actuation.

b. If a "realistic" calculation were performed, would water from the upper containment pool still be required? Would makeup to the suppression pool still be required?

Water from the upper containment pool (i.e., the SPMU "dump" function) would still be required for a "realistic" calculation.

Suppression pool makeup is not required for the DBA analyses performed in support of the proposed TS changes. As discussed in response to Questions 7a and 9, makeup to the suppression pool is required only for the special, beyond-design basis MSLB with bypass leakage time sensitivity studies.

c. If credit were not taken for the volume of equipment in the drywell pool, would the proposed limits be satisfied in a realistic analysis?

No. As stated in response to Question 15a, below, the equipment in the drywell pool displaces 863 ft^3. 863 ft^3 is equal to about 1.4 inches of suppression pool level. Therefore, if credit were not taken for this volume, and the initial suppression pool levels were at the proposed minimum levels, then the minimum post-accident suppression pool levels (using methods described in Section 4.3.1 of the submittal) would be 1.4 inches below the design minimum requirement to maintain 2 feet of LOCA vent submergence. However, realistically, the suppression pool levels would be expected to be maintained within the proposed operating ranges (5 2/3 inches for the gate installation evolution and 6 inches for the reactor cavity drain evolution). Thus the initial pool level could well be 1.4 inches higher than assumed in the analyses. In this case, the 2 foot submergence limit would be maintained without credit for the drywell equipment volume.

d. List any analysis assumptions that require operator action. Explain why the timing of these operator actions assumed in the analysis is reasonable. Are these operator actions consistent with Grand Gulf procedures?

The safety analyses supporting the proposed gate installation and MODE 3 reactor cavity drain evolutions credit manual operator action to control reactor water level below Level 8. This action is performed in accordance with the Grand Gulf Emergency Operating Procedures (EOPs). The action is necessary to maintain the post-accident suppression pool level at or above 14 feet 6 inches. This is the level required to maintain the "design" minimum vent submergence of 2 feet. In Section 4.3.7 of the submittal, results of analyses of a MSLB (3.54 ft^2) with Drywell bypass leakage ($A\sqrt{K}$) of 0.9 ft^2 is discussed. These analyses include time sensitivities to illustrate how much time is available before additional suppression pool makeup is needed to maintain 2 feet minimum vent submergence. The results for maximum Emergency Core Cooling System (ECCS) flow case show that if the operator controls water level in 7.5 minutes, external makeup to the suppression pool will not be required before 6 hours. If the operator controls level at 10 minutes, external makeup will not be required before 1 hour 23 minutes.

The above response times for operator action to control level are based primarily on long-term suppression pool inventory considerations for the MSLB with bypass leakage capability analysis. However, for short-term suppression pool level considerations, a finite amount of time is available to the operators to implement the vessel level control action. The time available varies depending on the break size and location, rate of reactor vessel depressurization, number of available ECCS pumps, and other operator actions (e.g., placing suppression pool cooling in service). A 10 minute response time ensures 2 feet of vent submergence (short term) under the most limiting assumptions for DBAs initiated during the gate installation evolution in MODES 1, 2, or 3 and the reactor cavity drain down evolution in MODE 3.

The 10 minute manual operator action specified in the analyses supporting the proposed gate installation/cavity drain evolutions is to monitor and maintain reactor vessel water level below L8 (the level band is discussed in response to Question 5a). This action requires simple control initiations, is proceduralized in the Grand Gulf EOP's, and can be performed from the control room. In addition, operators are trained (classroom and simulator) to take prompt action to limit filling the reactor vessel above the Level 8 limit.

The time that the plant will be in one of the configurations considered in the submittal is limited. To realize the benefits cited in Section 3.0, Attachment 1 of the submittal, gate installation will 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 (e.g., noble metal addition). In addition, the MODE 3 plant conditions required for initiating the proposed reactor cavity drain down include low (< 235 psig) vessel pressure and all rods in for > 3 hours. Since the plant will be undergoing a controlled cooldown, increased operator attention to vessel conditions (pressure, temperature and level) is expected. Considering the proceduralized operator actions and training, together with the above considerations, the 10 minute response time used in the safety analyses of the proposed evolutions is reasonable.

Manual operator action to provide makeup to the suppression pool from outside sources is considered only in the MODE 3 large break with bypass leakage capability analyses assuming a containment spray actuation with the reactor cavity drained. External suppression pool makeup is not considered in the analyses supporting the proposed gate installation. The MSLB with Max ECCS is the limiting case (see additional discussions in response to Question 7a). The operator action to provide external makeup water to the suppression pool is based on options available through the Emergency Procedures. Considering the conservative assumptions used in the bounding MSLB bypass leakage analyses (e.g., continuous operation of two spray loops), and the above considerations for operator action, a 1 hour 23 minute response time for this action is reasonable.

14. Describe the Appendix B program for the GOTHIC computer code at Grand Gulf.

GOTHIC was developed and is maintained by Numerical Applications, Inc. (NAI) under EPRI sponsorship and is fully qualified under the NAI QA program that conforms to the requirements of 10 CFR 50 Appendix B with error reporting in accordance with 10 CFR 50 Part 21. The code assessment program includes comparison to results from a wide range of analytic and experimental tests. Representative results from these comparisons are presented in the GOTHIC Qualification Manual. Grand Gulf is a member of the EPRI sponsored GOTHIC Users Group and as such is entitled to receive the GOTHIC code directly from NAI through the EPRI Software Center. The code is installed and verified at Grand Gulf prior to use in accordance with applicable software QA procedures. For procured software developed under an Appendix B program such as GOTHIC, these procedures 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.

15.

a. **What is volume used for equipment in drywell that is being credited for decreasing the drywell pool inventory?**

863 ft^3

b. **What equipment does this include?**

The credited 863 ft^3 reduction in drywell pool volume is the volume occupied by equipment between the weir wall and RPV pedestal wall below Elevation 117' - 4" (top of weir wall). The types of equipment are:

- Concrete equipment foundations
- Galvanized grating.
- Structural steel
- Steel pipe whip restraints
- Water filled piping (including recirculation system piping)
- HVAC duct work (only sheet metal thickness considered).

c. **What assurance is there that no changes will be made which affect (decrease) this volume?**

The above 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 analysis. Therefore, sufficient controls are in place to ensure that plant design changes that affect this equipment will receive Safety Analysis review for impact.

d. Describe any conservative assumptions made in computing the equipment volume.

The credited equipment volume represents equipment that can be readily accounted for and identified on plant design drawings. There are a number of items such as hand rails, stairs, drywell coolers, and miscellaneous steel items (angles, plates, duct and pipe hangers, etc.) that are not included. 86% of the credited volume (744 ft^3) consists of concrete equipment pedestals and recirculation piping

16. The use of GOTHIC to calculate mass and energy release uses a single volume for the reactor pressure vessel. Describe the assumptions made and the bench marking done to ensure that the GOTHIC reactor vessel model, including the mass and energy release calculations, is conservative for this application.

The reactor pressure vessel (RPV) model uses two volumes with multiple subvolumes. The vessel nodalization is shown in Figure 4.1, Attachment 4 of the February 25, 2002 submittal. The top portion of the vessel (steam dome region) is represented by Volume 3 in Figure 4.1. The remainder of the vessel is represented by Volume 1s. Volume 1s is subdivided into 12 subvolumes. Thus the reactor vessel effectively uses 13 volumes.

A subdivided GOTHIC volume is needed to capture the frothing effect that results in liquid leaving the break for the main steam line case. The model is subdivided into six vertical subvolumes to capture the frothing effect. These vertical divisions are made at heights that somewhat represent the elevations and heights of various vessel internal components. Horizontal divisions are also included to allow liquid from the frothing effect to 'fall' back. GOTHIC only allows phase flow in one direction at a time from one subdivided cell to another. Thus, one horizontal division was entered to separate the vessel volume into two vertical regions, the core region and the downcomer region. The top portion of the vessel is 'broken off' in a separate volume to force steam only flow through the break during appropriate times during the vessel blowdown. This also serves the purpose of modeling the effect of the steam separators and steam dryers, since they are not explicitly modeled. Note that, while the subdivisions in the GOTHIC RPV volume are loosely based on the general locations of various internal regions, the subdivisions exist only to capture the frothing effect and force fluid circulation and do not model separate internal regions or components. In accordance with this modeling style, all of the liquid in the RPV is assumed to be at the bottom of the vessel with equal liquid levels in both vertical channels at the beginning of the transient. The six vertical and one horizontal division in the lower RPV volume (Volume 1s) creates twelve GOTHIC subvolumes.

The GOTHIC RPV model described above was benchmarked to match the GE calculated blowdown mass and energy release rates from the MSLB DBA. This includes matching the blowdown liquid and steam flow and enthalpy rates shown in UFSAR Table 6.2-11 and Figure 6.2-19. As such, the vessel model incorporates all of the conservatisms and assumptions used in the approved GE methods. This model is also used to calculate the blowdown mass and energy releases for the low-pressure (235 psig) MODE 3 analyses.

17. Section 4.3.4:

a. **How is the reduction in long-term heat sink volume of 570 ft^3 determined?**

The reduction in long-term heat sink volume is determined using inventory balance methods as described in Section 4.3.1 of the February 25, 2002 submittal. For the current design, the long-term, post-accident suppression pool inventory, considering the initial suppression pool level at the TS Low Water Level, SPMU volume, and suppression pool entrapment (drawdown) volumes, is equal to 108,155 ft^3. For the proposed evolutions (gate installation or reactor cavity drain in MODE 3), the long-term, post-accident suppression pool inventory, considering the proposed increased initial suppression pool levels, reduced SPMU volumes, and entrapment (drawdown) volumes as described in Section 4.3.1 of the submittal, is equal to 107,588 ft^3. The difference is a net reduction in long-term pool inventory of 567 ft^3, or about 570 ft^3.

b. **Why is there a reduction in suppression pool volume if the adjustments of Section 4.3.1 of the submittal are made to the inventory in the suppression pool?**

Refer to Figure 2, page 16/19, Attachment 1 of the February 25, 2002 submittal for the suppression pool levels discussed in this response.

The reduction in long-term heat sink volume for the proposed evolutions compared to the current design is due to excess water in the pool for the current design. The current TS suppression pool low water level limit is 18 feet 4 1/12 inches (TS LCO 3.6.2.2). The basis for this LWL limit is a SPMU System design specification requirement that the initial pool level be 7 feet above the centerline of the top vent. With initial level at 18 feet 4 1/12 inches, the minimum long term post-accident suppression pool level is14 feet 7 inches, 1 inch above the required post-LOCA minimum 2 foot vent submergence level (14 feet 6 inches). The inventory requirements discussed in Section 4.3.1 of the submittal are designed to protect the 2 foot vent submergence requirement. Therefore, the minimum long-term suppression pool level for the proposed evolutions, using the adjustments described in Section 4.3.1, is equal to 14 feet 6 inches. The 1 inch difference is equal to about 570 ft^3 of suppression pool volume.

The suppression pool level is typically controlled at a level above the LWL limit near the center of the operating range. The proposed operating range that will be in effect during the time period that the UCP gates are installed is 18 feet 5 1/12 inches to 18 feet 9 ¾ inches. Therefore, while the above minimum post-accident suppression pool level and 567 ft^3 inventory reduction is based on suppression pool level at the proposed minimum (18 feet 5 1/12 inches), actual suppression pool level is expected to be > 1 inch above the proposed minimum, resulting in a minimum post-accident pool level > 14 feet 7 inches.

Similarly, the proposed operating range that will be in effect during the time period that the UCP level is below the current TS limit (23 feet 3 inches) is 20 feet 0 inches to 20 feet 6 inches. The proposed 20 feet 0 inch TS LWL limit includes a 1 inch margin for measurement uncertainty. The minimum post-accident suppression pool level and 567 ft^3 inventory reduction is based on a minimum suppression pool level of 19 feet 11 inches. Therefore, for this evolution the actual suppression pool level is also expected to be > 1 inch above the analyzed minimum level, resulting in a minimum post-accident pool level > 14 feet 7 inches.

c. **It is not clear how the referenced discussion of Humphrey issue 4.1 relates to less mass in the suppression pool. Isn't the mass accumulated in the drywell floor always part of containment calculations?**

The DBA LOCA containment response analyses assume that the ECCS pumps operate continuously with water flooding out the break and collecting in the drywell pool. The drywell pool is postulated to fill to the top of the weir wall, at which time it overflows into the suppression pool. Thus the suppression pool water is effectively mixed with the drywell pool.

The reference to Humphrey issue 4.1 is intended to show that a large reduction in suppression pool volume (mass) will not significantly affect the long-term suppression pool temperature. If this is the case, then the small (570 ft^3) pool volume reduction considered in the proposed gate installation evolution can be concluded to have an inconsequential effect on long-term pool temperature. Mr. Humphrey's concern was that, while the original DBA LOCA response analyses assumes that the drywell pool is mixed with the suppression pool, a relatively cool (135°F) drywell pool may in fact form and be isolated from the bulk of the suppression pool. The suppression pool heat capacity (mass or volume) assumed in the analysis would therefore be reduced while the remainder of the post-LOCA heat is transferred to the suppression pool. This could result in potentially higher long-term suppression pool temperatures and containment pressures.

To address Humphrey issue 4.1, GE reanalyzed this event and calculated the suppression pool temperature increase crediting only the post-LOCA suppression pool water mass (i.e., suppression pool isolated from the drywell pool). The inventory of the drywell pool was assumed to be thermally isolated from the suppression pool at a temperature of approximately 135°F and the remainder of the post-LOCA heat rejected to the suppression pool. This analysis showed that the effect on long-term suppression pool temperature is an increase in pool temperature of 10°F. GE stated that this increase was within identified margins in long-term suppression pool temperature response analysis. Isolating the drywell pool from the suppression pool results in a reduction in the mixed pool volume equal to the drywell pool volume, 49,261 ft^3 (3,029,552 lbm). Since this reduction was found to result in an increase in long-term pool temperature within identified margins, the 570 ft^3 (35,055 lbm) reduction in pool volume is bounded by this analysis and will therefore not result in a significant increase in pool temperature. The assessment of Humphrey Issue 4.1 is discussed in Section 6.2.1.1.3.3.2.1 of the GGNS UFSAR.

18. **Section 4.3.7: The GOTHIC analysis of the large break LOCA with steam bypass of the suppression pool assumes the operator controls vessel level to Level 8 within 7.5 minutes after LOCA initiation. Containment sprays are initiated at 10.75 minutes.**

a. **What is the basis for the 7.5 minutes?**

As discussed in response to Question 7a and 13d, the analysis of the large MSLB LOCA with suppression pool bypass included time sensitivities to illustrate how much time is available before additional suppression pool makeup is needed to maintain 2 feet minimum vent submergence. 7.5 minutes was one of the operator response times considered. This time is based on consideration of the volume of water entrapped in the drywell pool due to spillage from the break. Analyses assuming a 10 minute response time and both maximum and minimum ECCS were also considered. These times are considered reasonable estimates of actual operator response times for this event.

b. What is the basis for start of containment sprays at 10.75 minutes?

10.75 minutes is the lower nominal spray timer setpoint in the spray automatic actuation logic. The MSLB LOCA with suppression pool bypass analyses assume that the containment pressurization (due to the bypass leakage path) reaches the spray high containment pressure permissive setpoint (7.5 psig lower analytical value). The small break bypass leakage and the 2.5 ft^2 break with bypass leakage "Humphreys" analyses documented in UFSAR Section 6.2.1.1.5.5 assume a 13 minute timer setpoint, which is conservative for containment pressurization considerations. However, the 10.75 minute lower nominal setpoint is conservative for this analysis because earlier spray actuation decreases the time available for operator action to provide the external makeup to the suppression pool required for these events. Note that the delay in operator action to reduce ECCS flow to the vessel and control level significantly reduces the containment pressurization. Spillage of break water into the drywell depressurizes the drywell reducing the bypass leakage flow to the containment. In fact, a delay of 7.5 minutes would probably result in containment pressures remaining below the spray setpoint. For this reason, the 2.5 ft^2 break with bypass leakage "Humphreys" analysis referenced in the UFSAR credits operator action to control level before any liquid spills from the break. This assumption maximizes the containment pressurization. However, with no liquid spillage there is no water entrapment in the drywell pool and thus no long term suppression pool inventory concern because the spray entrapment volume due to the drained cavity is less than the drywell pool entrapment volume.

c. Is the start of containment spray a manual start? If so, how sensitive is the analysis results to the timing of the start of containment spray?

Containment spray start in this analysis is an automatic start. An earlier start could occur only by manual operator action and would proportionally reduce the time available to the operator to provide external makeup to the suppression pool. However, initiating containment sprays at < 10 minutes would divert two RHR/LPCI pumps from supplying flow to the reactor vessel. The 10.75 minute timer is designed to allow full ECCS injection to the vessel for at least 10 minutes following a LOCA to ensure adequate core cooling.

d. Is it necessary to make up to the suppression pool for any other LOCAs?

No.

e. What does it mean to require makeup to the suppression pool at a certain time? Has the level of the suppression pool reached 2-ft above the top vent at that time?

Yes.

f. Explain why these analyses are considered beyond the design basis.

The bypass leakage analyses are special calculations that evaluate the bypass leakage capability of the containment considering containment sprays and containment heat sinks (see discussion in UFSAR Section 6.2.1.1.5.5). The Grand Gulf licensed bypass leakage area is 0.9 ft^2 ($A/\sqrt{k}$) and is based on the results of a small (0.07 ft^2) steam line break with bypass leakage crediting one containment spray loop (at 13 minutes) and containment heat sinks. This analysis is described in Section 6.2.1.1.5.5 of the UFSAR. In response to Humphrey Issues 5.1 and 9.2, a sensitivity study of various break sizes from a small break to a DBA was conducted to determine containment pressurization prior to initiation of containment sprays. This study assumed a 0.9 ft^2 ($A/\sqrt{k}$) bypass leakage area and allowed for structural heat sinks and RPV level control to prevent liquid spillage from the break to maximize the drywell pressurization. The most limiting break was determined to be a 2.5 ft^2 steam line break. Results from this analysis show peak containment pressure at the time of containment spray initiation remains below the containment design pressure. Thus the small break bypass leakage case remains valid. Since the small break with bypass leakage analysis establishes the licensed bypass leakage area, this event is considered the design basis bypass leakage analysis. The 2.5 ft^2 steam break is a sensitivity analysis that supports the design basis small break results. The 3.54 ft^2 MSLB with 0.9 ft^2 bypass leakage event described in Section 4.3.7 of Attachment 1 of the submittal was developed specifically to maximize the demand on long-term suppression pool inventory for a LOCA in MODE 3 with the reactor cavity drained. This event assumes liquid spillage from the break for a finite time period together with containment pressurization and early spray initiation (see response to question 18(b)). As such, it is considered a special, beyond design basis event designed to create the most limiting suppression pool inventory requirement and identify limiting response times available to the operators for providing external makeup to the suppression pool.

19.

a. **Describe the transient calculations which determine the upper and lower inclined curves of TS Figure 3.10.9-1.**

As discussed on page 7/19 of the February 25, 2002 submittal, the curves in Figure 3.10.9-1 of the proposed Special Operations TS define limits for the UCP and suppression pool levels that will be in effect both during and after the reactor cavity drain down evolution. Maintaining the pool levels within the limits defined by these curves ensures that the combined water inventory in the suppression pool and UCP is sufficient to satisfy post-LOCA minimum suppression pool level requirements (2 feet of coverage above the top row of vents)

The cavity drain evolution will be initiated in MODE 3 with UCP filled to the current TS minimum level and the suppression pool level between the TS LWL and HWL operating limits (assuming that the UCP gate(s) have not been installed). The large in-place SPMU makeup volume, together with the reduced makeup volume requirement due to the reduced holdup volumes at MODE 3 conditions, results in excess water inventory in the containment when the drain evolution is entered

The upper and lower curves of Figure 3.10.9-1 are calculated for two drain down scenarios. Inventory requirements for these two scenarios bound all possible drain evolutions. Maintaining the combined pool inventories within the limits calculated for these scenarios ensures that the SPMU system remains operable (i.e., ensures that the combined SPMU volume and in-place suppression pool volume is sufficient to maintain 2 feet of vent submergence) during the drain down evolution. If the cavity gates (Gates 1 and/or 2) are installed prior to reaching the defined MODE 3 drain down conditions, the suppression pool level will be 1" above LWL. As discussed below in response to Question 19(b), using the current TS LWL is limiting for these calculations.

The bounding drain down scenarios are:

- UCP initially drained outside the suppression pool. The initial excess water in the UCP is drained outside the suppression pool. After the excess water has been drained (i.e., when the available makeup volume together with the in-place suppression pool inventory is just sufficient for SPMU operability), the remaining UCP inventory is drained directly to the suppression pool. This scenario, with suppression pool filled to the lower MODE 3 LWL (20 feet 0 inches), defines the lower curve in Figure 3.10.9-1.
- UCP initially drained to the suppression pool. The UCP inventory is drained directly to the suppression pool until the proposed suppression pool HWL limit is reached. The excess UCP inventory is then drained outside the suppression pool. The proposed suppression pool HWL limit is 20 feet 6 inches (per Section 4.3.1) allowable value, which protects the maximum analytical HWL limit of 20 feet 7 inches. This scenario defines the upper curve in Figure 3.10.9-1.

The equations and a sample calculation for a point on the upper and lower curves of Figure 3.10.9-1 are provided in Attachment 2 to this response.

b. **Page 7/19 of the February 25, 2002 submittal states that the current suppression pool low water level (LWL) is used in TS Figure 3.10.9-1 because it bounds the condition with the gates installed and minimum water level at 18-ft 5-1/12 inches. Please explain how it bounds the 18-ft 5-1/12 inch water level.**

   Suppression pool level at the current TS LWL of 18 ft 4 1/12 inches bounds the case with level at 18 ft 5 1/12 inches from the perspective of in-place suppression pool inventory. As discussed above, Figure 3.10.9-1 is entered when the required MODE 3 conditions for the reactor cavity drain evolution have already been met (i.e., 3 hours shutdown, 235 psig reactor pressure) and no water has been drained from the upper pools (UCP level is at 23 ft 3 inches). The minimum allowable suppression pool level is either the current TS LWL limit (18 ft 4 1/12 inches) or the 18 ft 5 1/12 inches limit proposed for the gate installation evolution if the gate has already been installed. Therefore, if the gate has been installed, the pool level is at or above 18 ft 5 1/12 inches and is already within the acceptable range of Figure 3.10.9-1.

c. **Also, please provide a sample calculation for a point on the inclined portion of the upper and the lower curves of TS Figure 3.10.9-1.**

   See Attachment 2

20. **What temperature is assumed for determining the density of water in MODE 3 for the Reactor Cavity Drained (MODE 3) situation? How sensitive is the final conclusion of sufficient available water to this assumption?**

The water temperature assumed for calculating water density is 401°F. This is the saturation temperature at the 235 psig vessel pressure analytical limit for the MODE 3 cavity drain evolution. Using saturated conditions is bounding as this provides the lowest possible water density and hence largest possible level shrink during the 235 psig to 0.0 psig vessel depressurization.

21. **Page 7/19 of Attachment 1 states that "containment loads have been evaluated and determined to be acceptable for suppression pool levels up to 20-ft 7-inches in MODE 3 when the reactor pressure is less than 235 psig." What load is limiting? What margin is available at 20-ft 7-inches?**

    This statement refers to the containment hydrodynamic loads evaluation documented in Section 4.3.2 of the submittal. These evaluations considered the impact of an increase in suppression pool water level of up to 21.25 inches above the current high water level limit, which corresponds to 20 feet 7 inches. The loads considered and the result of this evaluation are fully discussed in Section 4.3.2. No single containment load is limiting. Different hydrodynamic loads are limiting during various phases of the accident. For example, water jet loads peak during the first seconds of a LOCA as the drywell pressure rises while condensation oscillation loads occur after the LOCA vent clearing and the pool swell transient. The hydrodynamic loads evaluation does not quantify margin available at 20 feet 7 inches suppression pool level but demonstrates that the loads imparted with the 20 feet 7 inches level and reactor vessel pressure ≤ 235 psig are bounded by those from a DBA with the suppression pool filled to the current high water level limit.

22. **What is the basis for only one drywell bypass calculation with A = 0.07 ft^2 and $A/k^{1/2}$ = 0.9 ft^2? Why is this combination limiting for a small break at 235 psig? How was this determined?**

    The analyses supporting the proposed MODE 3 reactor cavity drain evolution consider a small (0.07 ft^2) break and two larger break (2.5 ft^2 and 3.54 ft^2) sizes with a bypass leakage path of 0.9 ft^2 $A/k^{1/2}$. The 0.9 ft^2 leakage path is the maximum licensed leakage and is the limiting size for all break sizes. GOTHIC analyses using the licensed bypass leakage path combined with small and 2.5 ft^2 (large) break sizes with the reactor vessel pressure at 235 psig show that the 2.5 ft^2 break with bypass leakage is the limiting combination for containment pressurization assuming one containment spray loop starting at 13 minutes.

23. **In Section 4.3.6, does the current drywell bypass calculation take credit for structural heat sinks? What conservatisms are used with the structural heat sink geometry and thermal properties?**

    Yes. The current drywell bypass calculation takes credit for structural heat sinks. As discussed in Section 6.2.1.1.5.5 of the GGNS UFSAR, the drywell bypass leakage analysis is performed to evaluate the bypass capability of the containment for small primary system breaks considering containment sprays and containment heat sinks as means of mitigating the effects on containment pressure of bypass leakage. The containment heat sinks are listed in Table 6.2-9 of the UFSAR.

    The GOTHIC analyses performed for the proposed MODE 3 reactor cavity drain evolution described in Section 4.3.6 also takes credit for structural heat sinks. This includes the small break with bypass leakage event. The heat sinks include the same types of structures considered in the current small break bypass leakage analysis described in UFSAR Table 6.2-9 (i.e., drywell structure, containment shell, miscellaneous structures and equipment). However, the heat sinks (number and geometry) are based on considerations of actual containment and drywell structures and geometry to reflect actual plant conditions. The credited miscellaneous structures and equipment represent equipment that can be readily accounted for and identified on plant design drawings. There are a number of items, such as hand rails, stairs, and miscellaneous steel items that are not included. The heat sink thermal properties use accepted values appropriate to the material type. The surface heat transfer model uses the Uchida correlation for condensing heat transfer. For the larger break bypass leakage cases crediting containment sprays, condensation heat transfer is conservatively neglected until spray flow is initiated. For the small break and MSLB DBA cases that do not credit spray, a conservative multiplier equivalent to the steam volume fraction in containment is applied the Uchida heat transfer coefficient.

24. **Section 4.3.6 states that the drywell bypass calculation was run without containment spray. Isn't this a typical assumption for drywell bypass calculations?**

The drywell bypass leakage calculations are typically performed assuming containment spray. As discussed in Section 6.2.1.1.5.5 of the GGNS UFSAR, the drywell bypass leakage analysis is performed to evaluate the bypass capability of the containment for small primary system breaks considering containment sprays and containment heat sinks as means of mitigating the effects on containment pressure of bypass leakage. The small break LOCA with steam bypass calculation discussed in Section 4.3.6, Attachment 1 of the February 25, 2002 submittal assumes no containment spray. This calculation demonstrates that, at the low reactor pressure MODE 3 conditions specified for the proposed reactor cavity drain evolution, the containment pressure does not increase above the containment design limit (15 psig) even with no pressure mitigation by the spray system.

25. **Attachment 4 of the February 25, 2002 submittal states that decay heat calculations were done using a "more realistic (less conservative [than May-Witt]) but bounding decay heat curve based on ORIGIN calculations." Please describe the decay heat model and how it is derived.**

As noted on page 5/19 of Attachment 4 of the submittal, the low pressure MODE 3 analyses starting at 3 hours after shutdown use a bounding decay heat curve based on ORIGEN calculations. This is the curve used in the Grand Gulf specific Emergency Procedure Guidelines (EPGs). The EPG decay heat curve is re-evaluated on a cycle-specific basis to ensure that a bounding decay heat curve is used for the current cycle. These decay heat curves are calculated using ORIGEN 2.1 with cycle specific inputs. These inputs include bundle specific powers, enrichments, and heavy metal mass. Conservatisms used in the calculations include:

- Enrichment and heavy metal mass information based on design data versus averaged as-built data.
- Depletion/decay schemes based on actual previous cycle and current projected cycle lengths. All previous outages (with one exception) are assumed 30 days long.
- No accounting for mid-cycle outages.
- No accounting for end-of-cycle power coast-down, the cycle is depleted at full power.

A decay heat curve for the end of current cycle (EOC 12) was calculated using the methods and conservatisms described above. The cycle 12 specific powers included allowance for the planned mid cycle 1.7% power uprate and were based on nominal exposures as this results in a slightly conservative (higher) decay heat level. This curve was compared to the curve used in the current EPGs. The current EPG decay heat curve (unchanged since cycle 5) was determined to be slightly higher and bounding. Therefore, EPG decay heat curve used in the MODE 3 GOTHIC analyses is the decay heat curve calculated for cycle 5. This curve is bounding for the current cycle and incorporates the conservative assumptions documented above. Comparison of this curve to the May-Witt curve shows that the May-Witt decay heat is only 0.24% greater than the EPG decay heat at 3 hours (180 minutes) after shutdown (starting point for the MODE 3 analyses) and this difference decreases with increasing time after shutdown.

26. **Figures 4.1 and 4.2 of Attachment 4 to the February 25, 2002 submittal show benchmarks of the short term drywell pressure and temperature as a function of time. For both the pressure and temperature curves, the GOTHIC results fall off more quickly after approximately one second than the FSAR calculations. Please explain this difference.**

    (Assume question is referring to Figures 4.2 and 4.3 of Attachment 4. This difference is discussed on page 5/19, Attachment 4 of the February 25, 2002 submittal.)

    The lower GOTHIC values of Drywell pressure (Figure 4.2) and temperature (Figure 4.3) during the period following the pressure peak are explained by the cooling effect of liquid droplets from the break. At one second, the MSLB fluid transitions from a steam to a liquid/steam mixture. In the GOTHIC model, the liquid component of the break flow disperses in drop form. These drops evaporate as they fall through the drywell to the drywell floor. The evaporation lowers the drywell temperature and pressure in the GOTHIC model. This effect is not included in the original GE methodology.

27. **Figure 4.3 of Attachment 4 of the February 25, 2002 submittal shows suppression pool long term temperature calculated with GOTHIC more conservative than the GE calculation. If a less conservative decay heat is used in the GOTHIC calculation as well as the same heat sink structures, why is the GOTHIC long term calculation more conservative?**

    (Assume question refers to Figure 4.4. Figure 4.3 of Attachment 4 of the February 25, 2002 submittal shows the short term drywell temperature comparison.)

    The GOTHIC calculation is a benchmark of the GE analysis of this event and as such uses inputs and assumptions consistent with the GE analysis. These include use of the May-Witt decay heat curve and no credit for containment heat sinks. (Page 5/19 of Attachment 4 describes the decay heat models used in the benchmarks and in the MODE 3 analyses.)

    - As shown in Figure 4.4, the long-term suppression pool temperature calculated using GOTHIC is conservative (higher) relative to the GE calculation. The peak suppression pool temperature predicted by GOTHIC is about 8°F higher than the GE value. The observed conservatism in the GOTHIC suppression pool temperature is primarily due to the methods used in GOTHIC to approximate the GE assumption of thermal equilibrium between the suppression pool and containment. Other conservatisms in the GOTHIC model that may contribute to this difference are:

    - Pump heat load from the operating RHR pump is added directly to the suppression pool water (the GE methods include this heat as heat added to the reactor vessel).

    - The GOTHIC suppression pool mass conservatively neglects the mass of water in the drywell vents (see inputs description on page 4/19 of Attachment 4). This results in a small reduction in suppression pool volume and heat capacity relative to the GE models.

28. Do the benchmark calculations of Figures 4.1, 4.2, 4.3 and 4.4 of Attachment 4 of the February 25, 2002 submittal use mass and energy release rates to the containment calculated by the GOTHIC model or the previous GE calculations?

The benchmark calculations documented in Attachment 4 use mass and energy release (blowdown) rates calculated by the GOTHIC model. This model has been benchmarked to the blowdown mass and energy from the GE calculations (see response to Question 16 for a description of the GOTHIC reactor pressure vessel model).

29. Describe the difference between the GE and the GOTHIC vent flow models. To what extent does this difference affect the benchmarking calculations?

The GOTHIC vent flow models include all phenomenon 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 can not 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. The GOTHIC vent flow model was benchmarked to the vent clearing times and drywell pressurization rates from the GE analyses of the MSLB and Recirculation Line Break with excellent agreement obtained for these important vent model parameters.

30. Figure 4.9 is a plot of containment pressure as a function of time for a main steam line break from a reactor pressure of 235 psig. This curve is used to demonstrate that the containment pressure remains below the set point for automatic containment spray actuation. What conservatisms are included in this calculation that would tend to increase the calculated containment pressure and provide confidence that the containment sprays will not be actuated?

The GOTHIC results in Figure 4.9 show a margin to the high containment pressure spray permissive setpoint (lower analytical value) of approximately 2 psi. Conservatisms included in this calculation that would tend to increase the calculated containment pressure include:

- Initial suppression pool level at the proposed lower analytical limit (19 feet 11 inches).
- Initial suppression pool volume that neglects the volume of water in the drywell vents.
- Conservative MODE 3 reactor vessel initial conditions. The initial conditions are calculated assuming a controlled vessel cooldown to the 235 psig analytical vessel pressure limit (versus 230 psig in the proposed TS) with liquid in the vessel at saturated conditions. This assumption does not credit the normally cooler (subcooled) liquid that would be supplied to the vessel during the cooldown evolution.
- Initial suppression pool temperature equal to 110°F (versus 95°F limit in the proposed TS).

- Upper Containment Pool (SPMU System water) initial temperature equal to 125°F (current TS limit).
- Limiting decay heat curve (see response to Question 25).
- Reactor vessel blowdown model benchmarked to the GE MODE 1 MSLB and incorporating the conservatisms and assumptions used in the approved GE methods. (The reactor vessel model is described in response to Question 16.)
- RHR heat exchanger performance based on the conservative GE model of the RHR heat exchanger.

31. The reactor pressure in MODE 3 is limited to 235 psig and the time after shutdown is limited to > 3 hours. What limits the reactor pressure to 235 psig? Is it hydrodynamic loads?

The 235 psig reactor pressure limit is derived from the analyses supporting the proposed MODE 3 reactor cavity drain. These analyses include the DBA LOCA and bypass capability studies, hydrodynamic loads analysis, and LOCA dose evaluation. Results from these analyses, taken together with the proposed increase in suppression pool level, establish the required MODE 3 reactor conditions (pressure, shutdown time). No single analysis establishes the 235 psig limit.

32. Describe how the allowance for the containment spray hold-up on equipment and structural surfaces is calculated.

This value (1,500 ft^3) is from the original GE Suppression Pool Makeup System Design Specification and is part of the original Grand Gulf licensing basis. The proposed evolutions described in the submittal do not impact this value.

33. Page 6/19 Attachment 1: For the case of <u>Gates Installed in MODES 1, 2, and 3</u>, the statement is made that "the value associated with the new suppression pool low water level is within the operating range of installed plant instrumentation. The operators can use the as-read value of pool level from installed plant instrumentation without any correction for instrument error or uncertainties." (a) Are uncertainties and instrument errors currently considered for the limits on suppression pool level? (b) Provide justification for not including instrument error or uncertainties either in the calculated required suppression pool level or as a correction to the reading made by the operator for this case.

It is noted that for the case of <u>Reactor Cavity Drained in MODE 3</u>, an instrument uncertainty is included (Page 7/19 of Attachment 1).

(a) The current suppression pool LWL limit is 18 feet 4 1/12 inches (per TS LCO 3.6.2.2). This is an analytical value based on a Suppression Pool Makeup System design requirement to maintain a minimum suppression pool level of 7 feet above the top vent centerline. As such, this limit includes no allowance for instrument error or uncertainty. The uncertainty associated with the narrow range instrument is 0.04 ft (approximately 0.5 inches). This uncertainty is currently accounted for in the surveillance procedure (Daily Operating Logs). This procedure's acceptance criteria for suppression pool level are ≥ 18.38 ft and ≤ 18.77 ft.

(b) The proposed TS suppression pool level limits are within the measurement band of the suppression pool narrow range level instrumentation used to enforce the current TS level limits. Therefore, the same instrument will be used to enforce the proposed limits and any considerations of instrument uncertainties and errors used to establish the current limits are also applicable to the proposed limits. In other words, implementation of the proposed changes will include the same (0.04 ft) instrument uncertainty as the current level limits, thereby protecting the analytical limit of 18 feet 5 1/12 inches.

For the case of Reactor Cavity Drained in MODE 3, a measurement uncertainty of 1 inch is applied to the analytical limits for the proposed suppression pool level range. This value was selected to bound the uncertainty when the proposed limits are implemented.

34. **Section 4.3.7: Analysis of large break LOCA with Steam Bypass of the Suppression Pool Bypass gives the results of a GOTHIC calculation which yields a peak containment pressure 0.5 psi below the containment design limit. What conservatisms are in the calculation?**

The large (2.5 ft^2) break LOCA with Steam Bypass leakage analysis that yields a peak containment pressure of 29.2 psia, or 0.5 psi below the design limit (15 psig, or ~29.7 psia), is the original GE analysis of this event at MODE 1 conditions. The GOTHIC analysis discussed in this section (Section 4.3.7 page 13/19) is an analysis of the 2.5 ft^2 break LOCA with Steam Bypass leakage initiated at the reduced MODE 3 reactor vessel pressure (235 psig). Initial suppression pool level is at the proposed high analytical limit (20 feet 7 inches) to maximize the bypass leakage and containment pressurization. This analysis shows a peak containment pressure of 26.93 psia, which is 2.77 psi below the 29.7 psia containment design limit. As discussed in response to Question 7(a), the bypass leakage analyses evaluate the bypass capability of the containment for primary system breaks considering containment sprays and heat sinks. As such, these analyses seek to push the containment pressure to the design limit to establish the spray setpoints (containment pressure and timer) and allowable bypass leakage area.

The GOTHIC analysis of the 2.5 ft^2 break LOCA with Steam Bypass leakage event shows that the peak containment pressure is below the corresponding MODE 1 result reported by GE. Therefore, the allowable bypass leakage area and spray setpoints established in the MODE 1 analysis remain bounding for the proposed MODE 3 conditions. Conservatisms in the GOTHIC bypass leakage calculation are the same as those discussed in response to Question 30 except for the initial suppression pool level and use of the RHR heat exchanger. Initial suppression pool level is at the proposed high analytical limit and suppression pool cooling is not considered. Conservatisms in heat sinks (geometry and thermal properties) are the same as those discussed in response to Question 23.

35. It appears that there is little or no margin in the volume of water available to the suppression pool in either the <u>Reactor Cavity Drained-MODE 3</u> case where a 1-inch uncertainty is included or the <u>Gates Installed (MODES 1, 2 3)</u> case where no allowance appears to be included for uncertainty. For example, credit is apparently needed for the volume of equipment in the drywell pool (863 ft^3) which is only approximately 1.5 % of the total entrapped volume.

GDC 50 requires sufficient margin in calculating the pressure and temperature following a LOCA. Discuss what margin in available water inventory exists, why this is acceptable or what margin might be added to ensure an adequate post-LOCA water inventory.

There is no explicit margin considered in the calculation of water inventories. This is consistent with current design basis methods.

The required initial volume of water in the suppression pool for the Reactor Cavity Drained-MODE 3 and Gates Installed (MODES 1, 2, and 3) cases are calculated using design basis methods for the Suppression Pool Makeup (SPMU) system described in UFSAR Section 6.2.7.1. Calculations of initial suppression pool volume, SPMU system "dump" volume and dump time, and suppression pool drawdown (entrapment) volumes are performed to ensure a minimum post-accident (after all drawdown volumes are filled) suppression pool inventory to maintain 2 feet of water coverage over the top of the top vent (equal to 14 feet 6 inches above pool bottom). These volume calculations do not explicitly consider calculation uncertainties but are performed using bounding assumptions and methods. These include:

- Consideration of all realistic water entrapment volumes, including water entrapped in all four main steam lines and the water required to fill the drywell pool.
- No consideration of alternate suppression pool makeup sources such as feedwater injection or HPCS/RCIC injection from the Condensate Storage Tank (CST).

These and other conservatisms and assumptions are discussed in the February 25, 2002 submittal and in responses to Question 13(a) and 15(c) (drywell pool volume and equipment) and Questions 5(b) and 9 (reduced reactor vessel entrapment volumes).

Multiple cases were considered in the safety analyses of the proposed evolutions to assure conservatisms in suppression pool inventory and peak containment pressure and temperature. For Gate Installation (MODES 1, 2, and 3), the existing DBA analyses described in Section 6.2 of the UFSAR were determined to remain bounding for both suppression pool inventory and post-LOCA pressure and temperature. For the proposed MODE 3 Reactor Cavity Drain evolution, analyses included a 3.54 ft^2 Main Steam Line Break (MSLB) with bypass leakage case assuming maximum ECCS, delayed operator action to control reactor water level, and an automatic containment spray initiation. This analysis, discussed in Section 4.3.7, Attachment 1 of the February 25, 2002 submittal, bounds the long-term suppression pool inventory requirements for a LOCA in MODE 3 with the reactor cavity drained. With respect to GDC 50, the multiple MODE 3 cases evaluated show substantial margin to the containment pressure and temperature design limits. The MODE 3 analyses incorporated the conservative assumptions used in the current licensed containment DBA analyses in UFSAR Chapter 6.2. These conservatisms are described in responses to Questions 13(a) and Question 30 and in Attachment 4 of the February 25, 2002 submittal.

36. **Contrast the amount of response time available prior to manual action for the operator in the control room between situation 1 and situation 2, where situation 1 is a loss of coolant accident (LOCA) in MODE 3 with water in the upper containment pool (UCP), and situation 2 is LOCA in MODE 3 without water in the UCP. Discuss the consequences of a failure on the part of the operator to take action under the two types of situations.**

The response to this question is based on the accident analyses performed by Entergy to support the proposed TS changes or bounding MODE 1 analyses.

No operator actions for situation 1 (LOCA in MODE 3 with UCP in normal configuration) are necessary and none are assumed in the safety analysis.

For situation 2 (LOCA in MODE 3 without water in the UCP), operator action to control reactor vessel water level to between Level 2 and Level 8 in accordance with the EOPs is assumed in the safety analysis (see response to Question 5 (a)). The action is necessary to demonstrate that the post-accident suppression pool inventory would be adequate to maintain level at or above 14 feet 6 inches. This is the level required to maintain the "design" minimum vent submergence of 2 feet. For purposes of the analysis, the time available for the operator to establish control of reactor water level varies depending on the break size, location, and assumed drywell bypass leakage. The most limiting event for suppression pool inventory is a MSLB (3.54 ft^2) with Drywell bypass leakage ($A\sqrt{K}$) of 0.9 ft^2 since containment spray actuation is expected. Analysis of this special capability study included time sensitivities to illustrate how much time is available before additional suppression pool makeup is needed to maintain 2 feet minimum vent submergence. The results for maximum ECCS flow show that if the operator controls water level in 10 minutes, external makeup to the suppression pool will not be required before 1 hour 23 minutes. For a MODE 3 MSLB LOCA without bypass leakage (DBA LOCA), the time available for operator action increases since containment spray actuation and the associated water holdup in containment is not expected to occur.

The safety analysis was patterned after that performed in support of the current suppression pool water level requirements and was intended to be a comparison to the current design basis. This design basis analysis was developed to ensure that adequate water level is available in the suppression pool to meet all water level requirements including maintaining 2 feet minimum vent submergence. The analysis developed in support of this change was intended to illustrate the differences in the holdup volumes and the resulting required makeup volumes between the proposal and the current design basis and to present a comparison of margins. The analysis assumes operator action. In the unlikely event that operators fail to properly execute the EOP actions, the plant response would maintain the key safety functions as described below.

For the MODE 3 MSLB LOCA event, failure of the operator to take any action would result in rapid filling of the reactor vessel to a level above the Main Steam Lines (MSLs) by the ECCS pumps. Reactor coolant would then flow out of the break into the drywell pool. The drywell pool would eventually fill to the top of the weir wall and overflow back to the suppression pool. In this scenario, reactor vessel blowdown to the drywell would be completed within a short period (much less than 10 minutes). Suppression pool level would eventually decrease to 13 feet 7 inches which is not adequate for the pressure suppression function of the containment. But, since the reactor vessel blowdown has already occurred, the pressure suppression function is no longer required.

Since the pressure suppression function is not required, the relevant and bounding safety function of the suppression pool water inventory becomes maintenance of adequate Net Positive Suction Head (NPSH) for the ECCS pumps. These pumps take suction from the suppression pool during a LOCA. As described in the GGNS UFSAR, ECCS pump NPSH has been shown to be sufficient for suppression pool levels as low as 12 feet 7.75 inches. This level is below the minimum suppression pool level of 13 feet 7 inches calculated for a MODE 3 LOCA assuming no operator action to control level and assuming the filling of the MSLs as described in the submittal.

The third safety function of the suppression pool is to ensure adequate suppression pool cooling capacity. The post accident pool heat capacity contains adequate margin such that the relatively small reduction in water inventory will not result in any significant increase in the long-term suppression pool temperature. Therefore, the relevant safety functions of the suppression pool are maintained even if operators do not control reactor water level and fail to align external makeup to the suppression pool as called for by the EOPs.

37. Describe any new operator actions required as a result of a LOCA in MODE 3 with the UCP drained.

There are no new operator actions required as a result of a LOCA in MODE 3 with the UCP drained. The operators would respond to control level in accordance with the EOPs. As discussed in Attachment 1 Sections 4.3.1 and 4.3.7 of the submittal, the action to control reactor vessel water level below Level 8 is already contained in the EOPs. Current guidance has the operator take 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.

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.

38. Describe any changes to any current operator actions covered by emergency operating procedures (EOPs) or abnormal procedures that may occur as a result of this amendment. Describe any evaluation of the EOPs for potential modifications.

The TS suppression pool limits are used for two of the entry conditions for the Containment Control EOP. The submittal changes these limits for the new scenario and thus the EOP will require changing. The changes will result in two sets of entry conditions which will depend on which TS is appropriate. TS 3.6.2.2 will be used for normal operations and proposed Special Operations TS 3.10.9 for operations in MODE 3 with the UCP drained.

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.

39. With regard to the minimum time available prior to mandatory manual action by the operator to control water level in the reactor for the two situations discussed above, describe information required by the control room operator to determine whether such operator action is required. What is the ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery?

As stated above, there are no mandatory operator actions required for either situation. For situation 2, action in accordance with the EOPs to control reactor vessel level is assumed in the safety analysis. This action already exists in the EOPs. EOP training, both classroom and simulator, is included in the ongoing Licensed Operator Training Program. There is no difference in the types of errors or the recovery from errors in this scenario than in any other event.

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.

40. Describe the difference in the level of difficulty with respect to any manual actions between the two situations discussed above. Under this amendment, what is the band of water level in the reactor that must be controlled by the operator?

The level of difficulty associated with the manual actions is the same and the actions are covered by existing EOPs. The band for reactor vessel level is the same for both scenarios and is specified in the EOPs. See response to Question 2 above for the specific range.

41. Describe any changes the proposed amendment will have on the operator training program, and provide the implementation schedule for making the changes.

The implementation process for TS changes dictates that all changes be reviewed by the Licensed Operator Training Staff to determine any effects on current training and the need for any new training. Once this amendment is approved, this review will take place.

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.

Attachment 2

To

GNRO-2002/00072

Sample Calculations

Attachment 2 - Response to Questions 19(a) and 19(c)
Calculation of UCP and Suppression Pool Level Curves for
Proposed Technical Specification Figure 3.10.9-1

Note: In the discussion below,

- *Figure 1 refers to Figure 1, Page 15/19, Attachment 1of the February 25, 2002 submittal.*
- *The criteria for SPMU System operability is that the combined water inventory in the suppression pool and UCP is such that post-LOCA suppression pool level (using methods described in Section 4.3.1 of the submittal) is 2 feet above the top row of vents.*

Equations and Inputs

Calculation of Transient SPMU (Dump) Volume

The transient SPMU volume when pool level is at or above the weir wall gate sill elevation (200' – 8 7/8" per Figure 1) is defined by:

$$V_{MU2}(h) = V_{MU1} - (A_C + A_{Sep} + A_{WW}) \times h \qquad L_{WW} \leq L_{CP} \leq L_{UCP} \tag{1.a}$$

where

V_{MU1} is the available SPMU System makeup volume with gates installed,
A_c is the cross-sectional area of the reactor cavity pool,
A_{Sep} is the cross-sectional area of the separator pool,
A_{WW} is the cross-sectional area above the separator pool weir wall.
(Values of these constants are given below.)

L_{WW} = pool level at weir gate sill (relative to pool bottom):

$$L_{WW} = (200' - 8\ 7/8") - (184' - 6\ 1/4") = 16.2188 \text{ ft} \quad \text{(cf. Figure 1)}.$$

h is the decrease in UCP level below the initial level and is given by

$$h = L_{UCP} - L_{CP},$$

where

L_{UCP} = normal (TS) UCP level, 23'-3" (23.25 ft),
L_{CP} = transient pool level during drain (ft).

After the UCP level decreases below L_{WW}, the makeup volume is given by:

$$V_{MU2}(h) = V_{MU1} - (A_C + A_{WW}) \times H_{MU} - (A_{Sep} \times h),\ \ L_{CP} \leq L_{WW} \qquad (1.b)$$

where

H_{MU} is the height of the water above the gate sill = (207' – 9 ¼") – (200' – 8 7/8") = 7.03125 ft.

Suppression Pool Inventory

The minimum required suppression pool inventory as a function of the UCP makeup volume [$V_{MU2}(h)$] is calculated based on maintaining the minimum containment water inventory (makeup and suppression pool) needed for SPMU operability for the MODE 3 reactor cavity drain conditions. For cases filling to the proposed 19'-11" analytical suppression pool level (L_{SP2}), the UCP pool required water inventory is:

$$V_{WR} = V_{MU2} + [(L_{SP2} \times A_{SP}) + V_v],$$

where

V_{MU2} is the required SPMU volume following drain down of the reactor cavity (= 12,333.5 ft^3, see Section 4.3.1 of the submittal),
V_v is the volume in the LOCA vents (= 2886.3 ft^3),
A_{SP} is the total suppression pool cross sectional area (= 7219.38 ft^2).

Substituting:

$$V_{WR} = 12{,}333.5 + [(19.9167 \times 7219.38) + 2886.3] = 12{,}333.5 + 146{,}672.5,$$

$$\mathbf{V_{WR} = 159{,}006\ ft^3}$$

For cases filling to the 20'-0" allowable (proposed TS) pool level (LWL(M3)), the minimum required containment pool inventory is:

$$V_{WR} = V_{MU2} + [(LWL(M3) \times A_{SP}) + V_v] = 12{,}333.5 + [(20.0 \times 7219.38) + 2886.3],$$

$$\mathbf{V_{WR} = 159{,}607.4\ ft^3}$$

SPMU operability is ensured so long as the total containment water inventory is $\geq V_{WR}$. Using this approach, the minimum required suppression pool inventory as a function of UCP inventory is given by:

$$V_{SPR}(h) = V_{WR} - V_{MU2}(h), \qquad (2)$$

and the required pool level is

$$L_{SPR}(h) = [V_{SPR}(h) - V_v]/A_{SP}, \qquad (3)$$

Values of constants in the above equations are as follows:

V_{MU1} = 28,072.2 ft^3 (see Section 4.3.1 of the submittal),
A_c = 1,296.0 ft^2,
A_{Sep} = 864.0 ft^2,
A_{WW} = 72.0 ft^2,

Drain Scenario 1: Initial UCP Drain Outside Suppression Pool

These calculations define the lower curve of Figure 3.10.9-1.

At the beginning of the drain down evolution there is an excess of water inventory in the containment. Therefore, a portion of the UCP can be drained outside the suppression pool without increasing the suppression pool level. When the combined UCP and suppression pool volume is equal to the minimum required volume, the remaining inventory in the UCP is drained directly to the suppression pool. The calculations proceed as follows:

1. For the transient UCP level decrease (h), calculate the available makeup volume using Equations 1a or 1b.

2. Determine the REQUIRED suppression pool inventory and level using Equations 2 and 3 and the appropriate value of V_{WR}.

3. Compare the required suppression pool level to the initial level (LWL or HWL). When the required level reaches the initial pool level, the remaining inventory in the UCP must be drained to the suppression pool to ensure the minimum containment water inventory requirement is met.

4. UCP drain continues to the suppression pool. The suppression pool fills to the 19' – 11" analytical limit or the 20' – 0" proposed TS limit when the UCP has been drained (final level is 16' – 2" above pool bottom).

The actual transient suppression pool level and volume are calculated as follows:

For initial level at LWL:

$L_{SPact} = L_{LWL}, \qquad L_{SPR}(h) \leq L_{LWL},$ (4.a)

$L_{SPact} = L_{SPR}(h), \qquad L_{SPR}(h) > L_{LWL},$ (4.b)

$V_{SPact} = (L_{SPact} \times A_{SP}) + V_v$ (4.c)

For initial level at HWL:

$L_{SPact} = L_{HWL}, \qquad L_{SPR}(h) \leq L_{HWL},$ (5.a)

$L_{SPact} = L_{SPR}(h), \qquad L_{SPR}(h) > L_{HWL},$ (5.b)

$V_{SPact} = (L_{SPact} \times A_{SP}) + V_v$ (5.c)

L_{LWL} = 18' - 4 1/12" = 18.34 ft (TS LCO 3.6.2.2),
L_{HWL} = 18' - 9 3/4" = 18.81 ft (TS LCO 3.6.2.2).

Equations 1 – 5 are used to calculate transient pool levels and inventories for this scenario. The resulting pool drain curves are shown in Figure 2. Four curves are calculated. Two beginning at the suppression pool LWL and filling to the proposed MODE 3 low water levels (19' – 11" analytical and 20'-0" allowable) and two beginning at the suppression pool HWL and filling to the proposed MODE 3 low water levels (19' – 11" analytical and 20'-0" allowable). The curve initiated at the suppression pool LWL limit and filling the suppression pool to the proposed analytical pool level (20 feet 0 inches) defines the lower bounding curve in proposed TS Figure 3.10.9-1.

Drain Scenario 2: Initial UCP Drain to Suppression Pool

These calculations define the upper curve of Figure 3.10.9-1.

In this scenario, the UCP is initially drained directly to the suppression pool until the suppression pool is filled to the proposed high water level limits, HWL(M3) = 20.5 ft (proposed TS limit) or $HWL(M3)_{MAX}$ = 20' – 7" = 20.5833 ft (analytical limit). The remainder of the UCP inventory is then drained outside the suppression pool.

The transient UCP makeup volume ($V_{MU2}(h)$) is given by Equations 1a or 1b. The transient suppression pool level and volumes are given by:

$$L^i_{SP} = L^{i-1}_{SP} + \frac{V^{i-1}_{MU2}(h) - V^i_{MU2}(h)}{A_{SP}} \qquad L^i_{SP} \le HWL(M3) \text{ or } L^i_{SP} \le HWL(M3)_{MAX}, \tag{6.a}$$

$$L^i_{SP} = HWL(M3) \qquad L^i_{SP} > HWL(M3),$$

OR (6.b)

$$L^i_{SP} = HWL(M3)_{MAX} \qquad L^i_{SP} > HWL(M3)_{MAX},$$

$$V^i_{SP} = \left(L^i_{SP} \times A_{SP}\right) + V_v. \tag{7}$$

where i denotes the current UCP and suppression pool volume and level and i – 1 the UCP and suppression pool volume and level at the previous calculated UCP level.

The calculation proceeds as follows:

1. For the given UCP level (h), calculate the available makeup volume using Equations 1a or 1b.
2. Calculate the resulting suppression pool level and inventory using Equations 6 and 7.
3. Compare the suppression pool level to the maximum allowable pool level (HWL(M3) or $HWL(M3)_{MAX}$). When the pool level reaches HWL(M3) or $HWL(M3)_{MAX}$, the UCP drain is directed outside the suppression pool.
4. UCP drain continues until the UCP has been drained to 16' – 2".

Equations 1 and 6 –7 are used to calculate transient pool levels and inventories for this scenario. Results of these calculations are shown in Figure 2. Four curves are calculated. Two beginning at the suppression pool LWL and filling to the proposed MODE 3 high water levels (20'-6" allowable and 20'-7" analytical) and two beginning at the suppression pool HWL and filling to the proposed MODE 3 high water levels (20'-6" allowable and 20'-7" analytical). The curve initiated at the suppression pool HWL limit and filling to the proposed 20'-6" Tech Spec HWL (HWL(M3)) is the upper bounding curve in the proposed TS Figure 3.10.9-1.

Sample Calculations

Lower Curve

This sample calculation uses the equations derived above for Scenario 1 to calculate a point at 19.0396 ft (19' – 0.48") UCP level on the lower curve of Figure 3.10.9-1.

L_{CP} = 19.0396 ft.

h = L_{UCP} = L_{CP} = 23.25 – 19.0396 = 4.2104 ft.

Since L_{WW} (16.2188 ft) $\leq L_{CP} \leq L_{UCP}$ (23.25 ft), equation 1a applies:

$V_{MU2}(h) = V_{MU1} - (A_C + A_{Sep} + A_{WW}) \times h,$

$V_{MU2}(h) = 28{,}072.2 - (1296.0 + 864.0 + 72.0) \times 4.2104,$

$V_{MU2}(h) = 18{,}674.587\ ft^3.$

The minimum required suppression pool inventory for cases filling to the 20'-0" proposed allowable (TS) pool level (LWL(M3)) is:

$V_{WR} = 159{,}607.4\ ft^3$

The minimum required suppression pool inventory as a function of UCP inventory is given by equation 2:

$V_{SPR}(h) = V_{WR} - V_{MU2}(h) = 159{,}607.4 - 18{,}674.613 = 140{,}932.81\ ft^3$

The suppression pool level is obtained using equation 3:

$L_{SPR}(h) = [V_{SPR}(h) - V_v]/A_{SP}, = (140{,}932.81 - 2886.3)/7219.38,$

$L_{SPR}(h) = 19.12$ ft.

This point (19.04 ft UCP level, 19.12 ft suppression pool level) is identified on attached Figure 2.

Upper Curve

This sample calculation uses the equations derived above for Scenario 2 to calculate a point at 19.25 ft (19' – 3") UCP level on the upper curve of Figure 3.10.9-1.

L_{CP} = 19.25 ft.

h = 23.25 – 19.25 = 4.0 ft.

Since L_{WW} (16.2188 ft) $\leq L_{CP} \leq L_{UCP}$ (23.25 ft), equation 1a applies:

$V_{MU2}(h) = V_{MU1} - (A_C + A_{Sep} + A_{WW}) \times h,$

$V_{MU2}(h) = 28{,}072.2 - (1296.0 + 864.0 + 72.0) \times 4.0,$

$V_{MU2}(h) = 19{,}144.2\ ft^3.$

Using results from the entire cavity drain down calculation, the UCP level at the previous (i – 1) calculation is 19.81 ft (19 feet 9.72 inches). The SPMU volume at this level is 20,394.12 ft^3 and the required transient suppression pool level is 19.876 feet. Thus,

$$V_{MU2}^{i-1}(h) = 20{,}394.12 ft^3 \qquad L_{SP}^{i-1} = 19.876 ft$$

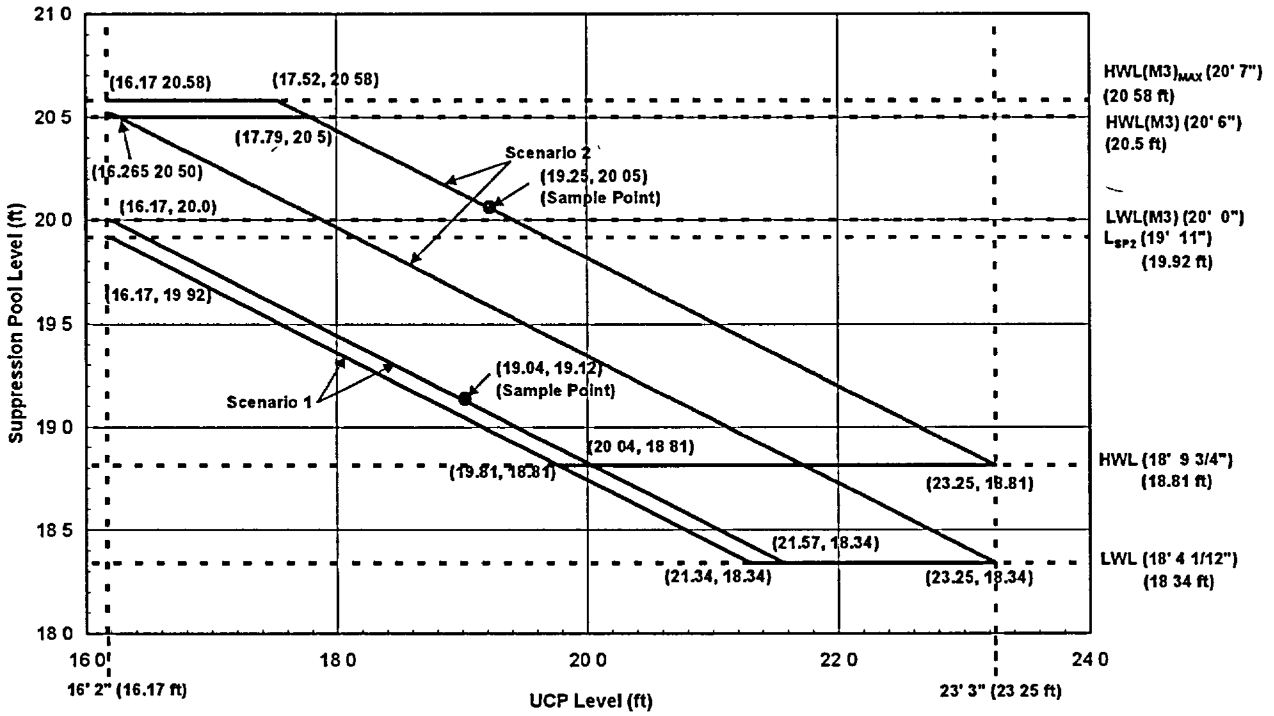
Substituting values into equation 6a gives the suppression pool level at 19.25 ft UCP level:

$$L_{SP}^{i} = 19.876 + \frac{20{,}394.12 - 19{,}144.2}{7219.38}$$

$$L_{SP}^{i} = 20.05 ft$$

This point (19.25 ft UCP level, 20.05 ft suppression pool level) is identified on attached Figure 2.

Figure 2


21 0
20 5
20 0
19 5
19 0
18 5
18 0
16 0
18 0
20 0
22 0
24 0
Suppression Pool Level (ft)
UCP Level (ft)
16' 2" (16.17 ft)
23' 3" (23 25 ft)
(16.17 20.58)
(17.52, 20 58)
(17.79, 20 5)
(16.265 20 50)
(16.17, 20.0)
(16.17, 19 92)
Scenario 2
(19.25, 20 05)
(Sample Point)
Scenario 1
(19.04, 19.12)
(Sample Point)
(20 04, 18 81)
(19.81, 18.81)
(23.25, 18.81)
(21.57, 18.34)
(21.34, 18.34)
(23.25, 18.34)
HWL(M3)MAX (20' 7")
(20 58 ft)
HWL(M3) (20' 6")
(20.5 ft)
LWL(M3) (20' 0")
LSP2 (19' 11")
(19.92 ft)
HWL (18' 9 3/4")
(18.81 ft)
LWL (18' 4 1/12")
(18 34 ft)

Attachment 3

To

GNRO-2002/00072

Revised Markup of Technical Specification Bases Pages

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.4 (continued)

the gates installed if the Suppression Pool Low Level limit is increased to 18 ft 5 ½ inches. (See Reference 3). The 31 day Frequency is appropriate because the gates are moved under procedural control and only the infrequent movement of these gates is required in MODES 1, 2, and 3.

INSERT A

SR 3.6.2.4.5

This SR requires a verification that each SPMU subsystem automatic valve actuates to its correct position on receipt of an actual or simulated automatic initiation signal. This includes verification of the correct automatic positioning of the valves and of the operation of each interlock and timer. As noted, actual makeup to the suppression pool may be excluded. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.6 overlaps this SR to provide complete testing of the safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a NOTE that excludes makeup to the suppression pool. Since all active components are testable, makeup to the suppression pool is not required.

REFERENCES

1. UFSAR, Section 6.2.
2. UFSAR, Chapter 15.
3. GNRO-2002/00011.

INSERT A to <u>SR 3.6.2.4.4</u>

The provision to allow gate installation in MODES 1, 2, and 3 results in isolating a portion of the SPMU System dump volume. This provision does not apply to the separator pool weir wall extension gates. These gates are not readily accessible with the upper containment pool at its required level. Supporting analyses have shown that increasing the minimum suppression pool level adequately compensates for water trapped by isolating the fuel storage and/or fuel transfer canal areas.