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GNRO-2002/00011

February 25, 2002

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

SUBJECT: Grand Gulf Nuclear Station, Unit 1
Docket 50-416
License Amendment Request New Special Operations LIMITING CONDITION FOR OPERATION Suppression Pool Makeup-MODE 3 (LDC 2002-006)

REFERENCE: Letter RS-01-250 from K. R. Jury (AmerGen Energy Company, LLC) to U. S. NRC, "Request for Amendment to Appendix A, Technical Specifications to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in MODE 3," dated November 16, 2001.

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations Inc., (Entergy) hereby requests amendment of Facility Operating License for Grand Gulf Nuclear Power Station (GGNS). Specifically, Entergy requests modification of the GGNS Technical Specifications 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 portion of the UCP while still in MODE 3, "Hot Shutdown," with the reactor pressure less than 230 pounds per square inch gauge (psig). Entergy also requests modification to the applicability of the UCP gates surveillance requirement (TS Section 3.6.2.4, "Suppression Pool Makeup (SPMU) System) to allow installation of UCP gates in MODE 1, "Power Operation," MODE 2, "Startup," or MODE 3. The proposed changes 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. This amendment request is similar to the license amendment request for the Clinton Power Station in the referenced letter.

Essential details and information to support this request are provided in the Attachments to this letter. Attachment 1 provides a description and justification for the requested TS changes. Attachment 1 also contains the evaluation for no significant hazards consideration, wherein it is concluded that, based on an evaluation of the proposed changes against the criteria of 10CFR50.92, no significant hazards consideration is involved. Attachment 1 also provides an evaluation against the 10 CFR 51.22 criteria for environmental considerations. The new Technical Specification pages showing the proposed changes are provided in Attachment 2, and the new Technical Specification Bases pages are provided for information in Attachment 3.

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Attachment 4 provides information related to the GOTHIC computer code used in support of this proposed change.

Since the proposed changes can provide significant reductions in outage critical path time, GGNS is respectfully requesting review and approval of this amendment by August 01, 2002. Once approved, the amendment will be implemented within 60 days. This would support scheduling of the activities before the outage such that planning for the outage can be finalized with the noted changes included in the outage scope. This letter contains no new commitments.

If you have any questions or require additional information, please contact Lonnie F. Daughtery at (601) 437-2334.

I declare under penalty of perjury that the foregoing is true and correct. Executed February 25, 2002.

Sincerely,

J Buck Edwards
for WAE

WAE/LFD/amt

attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes
3. Proposed Technical Specification Bases Changes
4. Information Related to the GOTHIC Computer Code

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ATTACHMENT 1

TO

GNRO-2002/00011

ANALYSIS OF PROPOSED TECHNICAL SPECIFICATION CHANGE

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-29 for Grand Gulf Nuclear Station, Unit 1 (GGNS).

Specifically, Entergy requests modification of the GGNS Technical Specifications (TS) to add a new Special Operations LIMITING CONDITION FOR OPERATION (Suppression Pool Makeup-MODE 3). This will allow installation of reactor cavity gate 2 in the Upper Containment Pool (UCP) and draining the reactor cavity pool portion of the UCP while still in MODE 3, "Hot Shutdown," with reactor pressure less than 230 pounds per square inch gauge (psig) and reactor subcritical > 3 hours. The purpose of the proposed change is to allow certain outage-related activities to commence while still in MODE 3. Entergy also requests the addition of a Note to the UCP gates surveillance requirement (TS Section 3.6.2.4, "Suppression Pool Makeup (SPMU) System) to allow installation of the reactor cavity gate in MODES 1, 2, or 3. Figure 1 shows the arrangement of the UCP and the UCP gates.

The proposed changes to the Technical Specifications are reflected in the annotated TS pages provided in Attachment 2. Associated changes to the TS Bases are indicated in Attachment 3. The proposed TS Bases changes are for information only and will be controlled by TS 5.5.11, "Technical Specifications Bases Control Program."

The next GGNS refueling outage is scheduled for the Fall of 2002. Entergy desires that this amendment be issued by August 1, 2002 to support work planning prior to the outage, since the proposed changes can provide significant reductions in outage critical path time.

2.0 PROPOSED CHANGES

The proposed change to the UCP gate Surveillance Requirement (SR 3.6.2.4.4) in TS Section 3.6.2.4 will allow installing the gate between the reactor cavity and fuel storage pool (gate 2) while operating in MODES 1, 2 or 3. TS SR 3.6.2.4.4 currently requires that the UCP gates be in the stored position or otherwise removed from the UCP with the unit in MODES 1, 2, and 3. The proposed change will add a note to this SR waiving the requirement provided that all UCP levels are maintained per SR 3.6.2.4.1 and that the suppression pool water level is maintained ≥ 18 feet 5 1/12 inches.

The proposed change to add a new Special Operations LIMITING CONDITION FOR OPERATION (LCO) will allow the draining of the reactor cavity pool portion of the UCP without having to declare LCO 3.6.2.2, "Suppression Pool Water Level" and LCO 3.6.2.4 "Suppression Pool Makeup (SPMU) System" not met. The specific change adds LCO 3.10.9 "Suppression Pool Makeup System-MODE 3", which stipulates new LIMITING CONDITION FOR OPERATION, APPLICABILITY, ACTIONS and SURVEILLANCE REQUIREMENTS for this Special Operation.

The proposed changes provide the controls necessary to ensure that the design inputs for the calculations performed to support operations with gates installed and the reactor cavity drained are continually met. The LIMITING CONDITION FOR OPERATION with the reactor cavity drained are set by the requirements of the Special Operation LCO. Variables, which must be periodically checked, are stipulated in the proposed SR. The proposed ACTIONS ensure that nuclear safety requirements are enforced at all times.

The new proposed TS Bases provide the basis for the new Technical Specification LCO and the added Note to SR 3.6.2.2.4. The proposed Bases provide sufficient information for operators to interpret the specification. The proposed Bases explain the reasoning for the NOTE allowing separate entry requirements and frequency requirements established for the SR.

3.0 BACKGROUND

A typical refueling outage requires that the plant be in MODE 4, "Cold Shutdown," before many of the critical path refueling floor activities can begin. For GGNS, these activities include installing gates in the UCP, draining the reactor cavity pool portion of the UCP, and decontaminating the cavity and Drywell head in preparation for Drywell head removal and vessel disassembly. In addition, GGNS plans to perform its first Noble Metal addition at the beginning of RF12 outage. The addition of this chemical requires that the plant be held in a MODE 3 status for the period of time of the injection. The injection and holding time lasts from 48 to 60 hours and would prevent the normal rapid progression into MODE 4.

To optimize outage scheduling, GGNS desires to install the UCP gates in MODES 1, 2, or 3 and to drain the reactor cavity pool portion of the UCP in MODE 3 without intentionally entering an LCO condition. By implementing the proposed changes to the Technical Specifications, gate installation and cavity drain activities can be started earlier than currently allowed. Allowing these activities to start earlier can result in a savings of up to six refueling floor critical path hours each outage, with savings of approximately 15 hours for outages involving Noble Metal addition.

Early draining of the UCP requires that the response of the plant be re-analyzed at the defined MODE 3 reactor conditions. The new analyses are performed using the GOTHIC code. GOTHIC is an advanced computer program used to perform transient thermal hydraulic analyses of multiphase systems in complex geometries. GOTHIC solves the conservation equations for mass, momentum, and energy for multicompartment, multiphase flow. The GOTHIC code has been benchmarked against test data and found to provide conservative results. GOTHIC has been previously used for Containment, high-energy line break and HVAC analyses at numerous nuclear power plants. The GOTHIC code has not been previously licensed for use at GGNS. Further discussion of the GOTHIC code and the GGNS model is contained in Attachment 4.

Gate installation and draining the UCP reduces the volume of water available to the Suppression Pool Makeup (SPMU) System. The GGNS Containment design depends on a minimum SPMU volume in the UCP, together with a minimum in-place suppression pool volume. The SPMU system consists of two independent lines which penetrate the UCP in the separator storage area. Each line has two motor operated valves, which are powered from the same electrical division, thus making each line an independent train. The system dumps by gravity a portion of the water from the UCP to the suppression pool to ensure Containment design requirements are met. A detailed description of this system can be found in the FSAR Section 6.2.7.

To compensate for the substantial reduction in UCP volume due to draining of the reactor cavity pool, the in-place suppression pool volume must be increased above the current high water level limit. There are three design features covered in the technical analysis section of this submittal that are affected by suppression pool level. Pool swell, caused by either a Loss of Coolant Accident (LOCA) or Safety Relief Valve actuation, is important due to the potential

damage caused to other equipment exposed to the swell. Weir wall overflow is of concern due to the potential flooding of the Drywell if this should occur. Horizontal (LOCA) vent submergence is also of concern due to the pressure suppression derived from forcing the steam out of the Drywell through the vents, which are submerged under water.

4.0 TECHNICAL ANALYSIS

4.1 Description of Current Requirements

TS Section 3.6.2.2 LIMITING CONDITION FOR OPERATION (LCO) requires the suppression pool water level to be maintained greater than or equal to 18 ft 4-1/12 inches and less than or equal to 18 ft 9-3/4 inches with the unit in MODES 1, 2 and 3. If the water level is not within limits, Required Action A.1 mandates the water level to be restored to within limits in 2 hours. Otherwise, the plant is required to be in MODE 3 in 12 hours and in MODE 4, "Cold Shutdown," in 36 hours.

TS Section 3.6.2.4 LCO requires two SPMU subsystems to be operable in MODES 1, 2, and 3. TS Surveillance Requirements (SR) confirm that the LCO is met. SR 3.6.2.4.1 requires verification that the UCP water level is greater than or equal to 23'-3". SR 3.6.2.4.2 requires verification that UCP water temperature is less than or equal to 125°F. SR 3.6.2.4.3 and 3.6.2.4.5 require verification of SPMU subsystem valve operability. SR 3.6.2.4.4 requires verification that all UCP gates are in the stored position or are otherwise removed from the UCP. If the water level is not within limits, Required Action A.1 mandates that the level be restored to within the limit in 4 hours. If the water temperature is not within limits, Required Action B.1 mandates that the temperature be restored to within the limit in 24 hours. If the gates are not in their stored position Required Action A.1 mandates that the level be restored in 4 hours. If the valves are inoperable, Required Action C.1 mandates that the SPMU subsystem be declared inoperable and restored to OPERABLE status within 7 days. If the level cannot be restored to within limits in 4 hours, or the temperature restored to within limits in 24 hours, or the SPMU system restored to operable status for any other reason within 7 days, the plant is then required to be in MODE 3 in 12 hours and in MODE 4 in 36 hours.

4.2 Bases for Current Requirements

The basis for the suppression pool maximum water level limit is to ensure that, following a LOCA, post-LOCA suppression pool swell loads, main steam safety/relief valve (S/RV) clearing loads, and other hydrodynamic loads are within design limits. In addition, maximum limits on the suppression pool water level ensure that the suppression pool will not overflow into the Drywell in the event of an inadvertent draining of the UCP into the suppression pool. The basis for the suppression pool minimum water level limit is to ensure that a sufficient amount of water is available to adequately condense the steam from Safety Relief Valve (S/RV) quenchers, main vents, or Reactor Core Isolation Cooling (RCIC) turbine exhaust lines. Minimum limits on the suppression pool water level also ensure an adequate emergency makeup water source to the Emergency Core Cooling System and provides a heat sink for the decay and sensible heat released during a reactor blowdown from S/RV discharges or from a LOCA.

The function of the Suppression Pool Makeup (SPMU) System is to transfer water from the UCP to the suppression pool after a LOCA. A portion of the water in the UCP is reserved for the SPMU system. Following accidents that result in depletion of the suppression pool inventory, the SPMU system transfers water (by gravity) from the UCP to the suppression pool. The basis

for the UCP minimum water level is to ensure that sufficient water volume is available to provide adequate post-accident suppression pool vent coverage of $\geq$ 2 ft above the top of the top row vents, to ensure that the suppression pool heat sink volume is adequate, and to ensure adequate net positive suction head (NPSH) for the Emergency Core Cooling System (ECCS) pumps.

4.3 Safety Analysis of Proposed Changes

The primary functions of the water level requirements for the UCP and the suppression pool are to ensure that the assumptions are met for post-accident water inventory. Sufficient water inventory in the suppression pool is required to provide ample coverage of the Drywell-to-Containment vents to ensure that the pressure suppression function of the Containment is performed. The suppression pool water inventory also ensures adequate net positive suction head (NPSH) for the Emergency Core Cooling System (ECCS) pumps. In addition, the long-term heat sink function of the suppression pool credits the volume transferred from the UCP. Each of these functions have been evaluated and determined to be acceptable during gate installation in MODES 1, 2, or 3 and draining of the UCP in MODE 3.

The proposed changes include:

- A revised operating range for the suppression pool water level (low water level limit) that will be in effect during the time period that the UCP gates are installed.

- A revised operating range for the suppression pool water level as a function of the UCP level that will be in effect during the time period that the UCP level is below the current TS limit (23 ft 3 inches). This revised operating range is above the current suppression pool high water level limit. The increase in suppression pool level has the potential to impact post-LOCA Containment pressure/temperature (thermal-hydraulic) response and hydrodynamic loads. An evaluation of the thermal-hydraulic and hydrodynamic load impacts is provided below. An analysis for the steam line break LOCAs with steam bypass of the suppression pool has special considerations and is evaluated specifically. Miscellaneous considerations are also discussed.

- Requirements that the reactor steam dome pressure be <230 psig and the reactor be subcritical for >3 hours will be in effect during the time period the UCP level is below the TS limit (23 ft 3 inches). These requirements ensure that the reactor conditions are bounded by the supporting analyses. The proposed reactor pressure limit includes a 5 psi margin to the analytical limit (235 psig) for measurement uncertainty.

4.3.1 Post-Accident Vent Coverage

The initial minimum suppression pool water level ensures adequate vent coverage during the first minutes of a LOCA. As the ECCS systems draw down the level in the suppression pool and inject water into the reactor vessel, the spillage through the line break forms a pool in the bottom of the Drywell (the "Drywell pool"). This drawdown results in the suppression pool level being reduced until the water from the break, collecting in the bottom of the Drywell, reaches an elevation that overflows the Drywell weir wall and returns the inventory to the suppression pool. The volume of water needed to flood the Drywell to a level above the weir wall, combined with the other entrapped water volumes described below, are considered in the long-term accident

analysis. The sizing of the UCP accounts for the minimum suppression pool level (after filling the required hold-up volumes) to assure that the suppression pool water level will provide at least 2 feet of coverage above the top row of vents.

The basis for the required makeup volume considers four primary entrapped volumes.

1. The free volume inside and below the top of the Drywell weir wall (the Drywell pool);

2. The added volume required to fill the reactor pressure vessel from a condition of normal power operation to a post accident complete fill of the vessel, including the top dome;

3. The volume in the steam lines out to the inboard main steam isolation valve (MSIV) on three lines and out to the outboard MSIV on one line;

4. Additional allowance for Containment Spray (CS) hold-up on equipment and structural surfaces.

These four volumes are considered in the design basis makeup requirements, but contain margins when considering an accident during the proposed operating conditions. The impact on each of these volumes and the resulting required suppression pool volume for each of the proposed operating conditions are described below.

Gate Installed (MODES 1, 2, 3)

The volume of the Drywell is fixed and must be considered for a design-basis Reactor Recirculation System line break in all plant modes. However, the design volume conservatively neglects the volume reduction due to equipment. As a result, there is a small margin in the analysis associated with this volume.

The Emergency Operating Procedures (EOPs) direct the operators to maintain reactor vessel water level below Level 8 (i.e., high reactor vessel water level). Therefore, the level in the vessel from level 8 to the top of the dome is volume that does not need to be considered. Since the steam lines may fill prior to operators taking action to reduce ECCS flow, the volume needed to fill the steam lines is included. The assumption that the reactor vessel is flooded to level 8 rather than the top of the dome is a change from the current licensing basis entrapment volumes for calculating required suppression pool make-up and is applicable for the scenarios in this amendment only.

The allowance for CS hold-up on equipment and structural surfaces is unchanged. Additional CS hold-up volume is included for filling the pools isolated by gate installation.

Reducing the required makeup volume by the margin in the Drywell pool due to equipment and by the volume needed to fill the reactor vessel from level 8 to the top of the reactor vessel steam dome and including the additional CS hold-up volume results in a net reduction in the total entrapped volume to 56,377.3 cubic feet.

The makeup that will continue to be available from the UCP following gate installation is 28,072.2 cubic feet. The net difference between the total entrapped volumes and the available make-up from the UCP is 28,305.1 cubic feet, which represents the drawdown volume that is required to be available in the suppression pool. The current suppression pool low water level

(LWL) limit provides 27,724.4 cubic feet of drawdown volume to maintain 2 feet of vent coverage post-accident. The difference between the required drawdown volume of 28,305.1 cubic feet and the available volume of 27,724.4 cubic feet is 580.7 cubic feet. This is the volume that must be provided above the current low water level to ensure 2 feet of vent coverage with the reactor cavity gate installed. To provide 28,305.1 cubic feet of available water volume, the suppression pool water level must be approximately 1 inch higher than the current low water level limit of 18 feet 4 1/12 inches. This results in a new minimum water level limit of 18 ft 5 1/12 inches providing a suppression pool level operating range of 4 2/3 inches, which is reduced from the current 5 2/3 inch range. Even though this operating range is slightly reduced, there is negligible impact on plant operations with this smaller operating band during operations with the gate installed. Figure 2 shows the current and proposed suppression pool water level limits as well as other important suppression pool elevations. Figure 1 shows the arrangement of the UCP and important UCP water levels and elevations. The value associated with the new suppression pool low water level is a nominal value since the new limit 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 correction for instrument error or uncertainties.

Reactor Cavity Drained (MODE 3)

Similar to the previous condition (gate installed), the volume of the Drywell is fixed and must be considered for a design-basis reactor recirculation system line break in any plant mode. However, the design volume conservatively neglects the volume reduction due to equipment. As a result, there is a small margin in the analysis associated with this volume.

The Emergency Operating Procedures (EOPs) direct the operators to maintain reactor vessel water level below Level 8 (i.e., high reactor vessel water level). In addition, the vessel thermal-hydraulic conditions in MODE 3 are bounded by MODE 1 design conditions. The vessel refill design volume includes allowances to compensate for the change in the density of the vessel liquid (level shrink) due to post-LOCA vessel depressurization and for collapse of steam voids. For the defined MODE 3 vessel conditions ($\leq$ 235 psig pressure), there is essentially no voiding below the water level and the level shrink is substantially reduced. Therefore, the level in the vessel from level 8 to the top of the dome and the volume to compensate for steam voids and level shrink from MODE 1 conditions is volume that does not need to be considered. Since the steam lines may fill prior to operators taking action to reduce ECCS flow, the volume needed to fill the steam lines is included. The assumption that the reactor vessel is flooded to level 8 rather than the top of the dome is a change from the current licensing basis entrapment volumes for calculating required suppression pool make-up and is applicable for the scenarios in this amendment only.

The CS is not required and will not automatically start for a large line break LOCA at the defined MODE 3 conditions. A GOTHIC analysis of a large (main steam line) break LOCA at MODE 3 conditions calculated a peak containment pressure of 5.94 psig (Figure 4-9), which is below the lowest CS actuation pressure (7.5 psig analytical). Therefore, the CS hold-up volume need not be considered.

Reducing the required makeup volume by the margin in the Drywell pool due to equipment, by the volume needed to fill the reactor vessel at MODE 3 conditions and from level 8 to the top of the reactor vessel steam dome, and by the spray holdup volume reduces the total entrapped volumes to 51,361.8 cubic feet.

The makeup that will continue to be available from the UCP following draindown is 12,333.5 cubic feet. The net difference between the total entrapped volumes and the available make-up from the UCP is 39,028.3 cubic feet, which represents the drawdown volume that is required to be available in the suppression pool. The current suppression pool high water level (HWL) limit provides 31,134 cubic feet of drawdown volume to maintain 2 feet of vent coverage post-accident. The difference between the required drawdown volume of 39,028.3 cubic feet and the available volume of 31,134 cubic feet is 7,894.3 cubic feet. This is the volume that must be provided above the current high water level to ensure 2 feet of vent coverage with the reactor cavity portion of the upper pool drained. To provide 7,894.3 cubic feet of available water volume, the suppression pool water level must be approximately 13.25 inches higher than the current high water level limit of 18 feet 9 ¾ inches. This results in a new minimum water level analytical limit of 19 ft 11 inches. Providing an operating range of 6 inches, which is slightly increased over the current 5 2/3 inch range, and including 1 inch for measurement uncertainty, the new suppression pool level range for operation in MODE 3 with the cavity drained and reactor pressure less than 235 psig and subcritical for > 3 hours is $\geq$ 20 ft 0 inch and $\leq$ 20 ft 6 inches. With measurement uncertainty, the maximum suppression pool water level analytical limit is 20 feet 7 inches, which is 1 foot 9 ¼ inches (21.25 inches) higher than the current high water level limit (see Figure 2). 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.

The following table summarizes the available and required make-up volumes for this operating condition.

Total post-accident entrapped volumes:	+ 51,361.8	cubic feet
Make-up still available from the upper pool:	- 12,333.5	cubic feet
Make-up available from suppression pool when level is at HWL	- 31,134	cubic feet
Additional water volume required:	+ 7,894.3	cubic feet

Reactor Cavity Drain Evolution (MODE 3)

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

Transient UCP and suppression pool water inventory requirements have been calculated to bound all possible drain evolutions. The drain evolutions consider initial suppression pool level at the TS LWL limit and HWL limit. If the cavity gate is installed prior to reaching the MODE 3 drain down conditions, the suppression pool minimum level will be 1 inch above LWL. However, using the current TS LWL limit bounds the condition with gates installed and minimum pool level at 18 foot 5 1/12 inches. The result of this evaluation is the bounding UCP and suppression pool level curves incorporated into the proposed Special Operations LIMITING CONDITION FOR OPERATION (LCO) 3.10.9 as Figure 3.10.9-1 (see Attachment 2). Maintaining pool levels within the limits defined by this figure ensures that the suppression pool water level will provide at least 2 feet of

coverage above the top row of vents during and after the reactor cavity has been drained in MODE 3.

4.3.2 Hydrodynamic Loads

The proposed Special Operations LCO requires raising the water level in the suppression pool above the current TS HWL limit in MODE 3 with the reactor pressurized. This has the potential to increase the hydrodynamic loads from both LOCA and S/RV actuations. Evaluations were performed on the hydrodynamic loads in the Containment due to a primary system pipe break. These evaluations considered the impact of an increase in suppression pool water level of up to 21.25 inches (1 foot 9 ¼ inches) above the current high water level limit. The evaluations show that the hydrodynamic loads imparted with the revised water level and reactor pressure below 235 psig will be bounded by those from a design basis accident with the suppression pool filled to the current high water level limit. For this evaluation, the term Design Basis Accident (DBA) is defined as the Main Steam Line Break (MSLB). The term LOCA is used to refer to the full spectrum of break sizes and initial conditions. Each of the Containment loads, identified in the GGNS Final Safety Analysis Report (FSAR) Appendix 6A and 6D (Reference 1 and 2) is considered.

The Containment loads generated during the first part of a LOCA are primarily a function of the Drywell pressure rise and secondarily a function of the suppression pool water level, temperature and other parameters. A GOTHIC analysis of a LOCA, with the primary system at 235 psig following 3 hours of post-shutdown decay and with the suppression pool water level at 1 foot 9 ¼ inches (21.25 inches) above the high water level, calculated a peak Drywell pressure of 10.8 psig. This compares with a peak Drywell pressure of about 22 psig for the Main Steam Line Break DBA with the suppression pool at the high water level limit (Reference 3).

Water Jet Loads

As the Drywell pressure rises during the first seconds of a LOCA, the water standing in the Drywell to Containment (LOCA) vents will be forced into the Containment (vent clearing transient). Water jets from the LOCA vents will impose impingement loading on the Containment wall. It has previously been determined that for a DBA, these loads are small compared to other loads that are imparted later in the transient (Reference 2). These loads are primarily a function of the Drywell pressure. For the low pressure LOCA, the loads will be smaller than the DBA loads due to the reduced Drywell pressure.

LOCA Air Bubble Loads

As pressurized Drywell air is forced through the vents and rises through the suppression pool, differential pressures are imposed on the weir wall, Drywell wall, Containment wall and basemat. These pressure differentials arise from the Drywell to Containment pressure differential, the low pressure in the annulus due to the high velocity flow and the local pressure variation due to the bubble formation in the suppression pool. The peak Drywell to Containment pressure differential and the peak vent flow are lower for the low pressure LOCA than for the DBA and the associated loads will be small for the low pressure LOCA. As discussed below, pool bubble dynamics in the suppression pool are expected to be less severe for the lower pressure LOCA and therefore produce lower pressure differentials.

Pool Swell Drag and Impact Loads

After the LOCA air bubble clears the first row of vents, the bubble lifts the pool surface. The bubble rises through the rising pool surface, eventually breaking through and forming froth at the top of the pool. There are impact loads on equipment and structures that are initially above the pool surface and drag loads on equipment and structures within the pool swell zone. The impact loads are a function of the pool surface velocity, which is a function of the air flow through the vents. The bubble grows until the pressure inside the bubble comes into equilibrium with the pool pressure. An increase in pool level will tend to increase the bubble pressure. However, for the low pressure LOCA, the Drywell pressure driving the bubble is smaller than the Drywell pressure during a DBA and the vent flow rates are smaller, which reduces bubble pressure, size and rate of growth.

Testing has shown that pool swell velocity is substantially reduced (by about a factor of 2) if there is venting through only one row of vents as opposed to two rows of vents (Reference 4 page 3-3). The Mark III pool swell load definition is based on the even higher loads developed from all three rows of vents clearing. For the low pressure LOCA, the GOTHIC analysis predicts that most of the venting will occur through the top row of vents with only a small amount passing through the second row of vents and none through the lower vents. This will reduce the pool swell velocity by at least a factor of two from the design load definition. The combination of the significantly lower Drywell pressure and clearing only the top row of vents significantly reduces the LOCA bubble size and the pool swell velocity. The lower swell velocity is also supported by GE Tests (Reference 5) which indicate that for similar break conditions, the peak swell velocity decreases with increased vent submergence.

Fallback Loads

After the bubble breaks through the pool surface, water will fall back to the pool imposing impact and drag loads on equipment and structures. Since the pool swell in the low pressure LOCA is bounded by the DBA, the maximum velocity of the falling water, impact loads and drag loads will be smaller in the low pressure LOCA.

Froth Impingement and Drag Loads

When the bubble breaks through the pool surface, the release of air from the pool creates a froth that can impinge and drag on structures and equipment and, in particular, on the Hydraulic Control Unit (HCU) floor. Since the initial bubble volume, the pool swell and the vent flow rates are all smaller in the low pressure LOCA than in the DBA, there will be less froth and the maximum froth level will be lower in the low pressure LOCA. Therefore, these loads are all bounded by the DBA.

Condensation Oscillation and Chugging Loads

Condensation oscillation can occur in the low pressure LOCA and DBA as well as the intermediate and small break accidents when the vent flows are in the critical range. When the steam mass flow through the top row of vents falls below 10 lbm/ft^2-sec, the condensation oscillation transitions to an erratic chugging mode. Condensation oscillation and chugging loads are not dependent on peak Drywell pressures and therefore may not be attenuated for a smaller or lower pressure LOCA. Condensation oscillation loads are independent of vent submergence (Reference 6) and therefore are not increased by these proposed changes. The impact of

increased vent submergence on chugging loads was previously reviewed by the NRC as part of the resolution to Humphrey Concern 19.1 (Reference 7). The conclusion was that the margins inherent in the chugging loads definition were more than adequate to accommodate increased vent submergence of up to 4.5 ft.

The suppression pool contains partially submerged piping that will be submerged for more of their area due to the increased suppression pool height. Partially submerged structures include piping such as ECCS relief valve discharge lines, test return lines, and minimum flow lines that enter the suppression pool vertically and are submerged for part of their length. Increasing the suppression pool level increases the area on which loads, such as condensation oscillation and chugging, can act. Based on a review of the stress analysis for the partially submerged piping, it is expected that the piping stresses will remain within the ASME code allowables with the increase in suppression pool level. This is primarily because the condensation oscillation and chugging loads do not represent the limiting loads for the partially submerged piping in the suppression pool.

Drywell Depressurization Loads

Between 100 and 600 seconds after a Main Steam Line Break (MSLB) DBA, the ECCS systems refill the vessel to the elevation of the break and, assuming the operators do not throttle ECCS, relatively cold ECCS fluid spills out of the break. This spillage causes the steam in the Drywell to rapidly condense, reducing the Drywell pressure. This causes inward loads on the Drywell wall. When the Drywell pressure is sufficiently lower than the Containment pressure, the suppression pool can be drawn backwards through the vents into the weir annulus and up through the annulus into the Drywell. The rapid reverse flow introduces potential jet impingement, impact, and drag loads on the weir wall and structures above the weir annulus. All of these loads are primarily a function of Containment pressure at the time of depressurization. The design basis for these loads assumes that Drywell vacuum breakers are non-functional and that the Containment temperature corresponds to a suppression pool temperature that maximizes the Containment pressure. For the low pressure LOCA, the energy deposited to the suppression pool will be less than in the DBA and so the suppression pool temperature will be lower. This will give lower Containment pressures and therefore lower Drywell depressurization loads.

Suction Strainer Uplift

Uplift loads on the ECCS/RCIC suction strainers during a DBA include dead weight and buoyancy forces, inertia loads due to seismic events, and hydrodynamic and inertia loads induced by S/RV actuation and LOCA venting to the pool. The maximum LOCA loads occur during chugging. The DBA chugging loads are primarily a function of bubble growth rate which will be smaller during a lower pressure LOCA. Further, DBA chugging loads were determined in Reference 7 (resolution of Humphrey Concern 19.1) to be acceptable for increased vent submergence up to 4.5 feet. Therefore, the DBA chugging loads are expected to bound the low pressure LOCA chugging loads. The phenomenon governing S/RV actuation loads are similar to that for chugging. The impact of increased pool depth will be similar to that for chugging and therefore the DBA S/RV loads are bounding for the low pressure LOCA. Weight and buoyancy loads and loads due to seismic events are independent of pool depth and are therefore not affected by the proposed changes. Thus, the combined uplift loads on any segment of the strainers for the low pressure LOCA are bounded by the DBA such that the strainer will remain on the basemat during the low pressure LOCA events.

Safety/Relief Valve Lift Loads

Hydrodynamic loads from S/RV actuation are partially dependent on discharge leg submergence. The loads, however, are far more dependent on vessel pressure. The impact of increased suppression pool levels up to five feet over normal suppression pool high water level on S/RV loads was previously addressed for resolution of Concern BNL-2 (Reference 7, 9). This design basis analysis shows that the S/RV Discharge Line (S/RVDL) thrust loads which would result from S/RV actuation at elevated suppression pool levels are within the upset allowable stresses. With vessel pressure of <235 psig, the loads from an S/RV lift will be significantly less than the design values.

4.3.3 ECCS NPSH Requirements

The ECCS pumps, including the Low Pressure Coolant Injection (LPCI), High Pressure Core Spray (HPCS), and Low Pressure Core Spray (LPCS) system pumps, have been analyzed for NPSH requirements in FSAR Appendix 6E. The analyses are performed assuming 212°F suppression pool temperature (clean strainer) and 185°F suppression pool temperature (strainer fully loaded), pump design runout flows, and atmospheric Containment pressure. These analyses show that adequate NPSH is available with the suppression pool level at the minimum drawdown level, 14.5 feet above the bottom of the suppression pool. This pool level is also sufficient to eliminate concerns such as vortexing, flashing, and cavitation during a LOCA. The proposed changes to the suppression pool and UCP levels ensure that the minimum suppression pool drawdown level (14.5 feet) is protected. Therefore, there are no concerns regarding ECCS pump NPSH requirements as a result of these changes.

4.3.4 Long Term Heat Sink

The suppression pool volume provides a long-term heat sink for the decay and sensible heat released following a LOCA. The long-term suppression pool volume, considering makeup by the SPMU system, is reduced by about 570 cubic feet due to the proposed changes. This volume reduction could impact the long-term suppression pool temperature and consequentially, the Containment air pressure and temperature. The impact of a decreased long-term Containment pool volume was previously addressed as part of the resolution to Humphrey Concerns Issue 4.1 (Reference 7). An analysis was performed to evaluate the maximum effect on bulk suppression pool temperature following a DBA LOCA with the suppression pool thermally isolated from the Drywell pool. Drywell pool isolation decreases the long-term heat sink by 49,216 cubic feet. This analysis showed that the maximum increase in bulk pool temperature is only 10°F. This increase was well within identified margin and bounds operations with the reactor cavity gate installed, which results in a much smaller reduction in long-term suppression pool volume.

For the proposed Special Operations LCO, an analysis was performed using GOTHIC considering the MSLB DBA with the reactor cavity drained, reactor pressure equal to 235 psig, and initial suppression pool temperature equal to 110°F (TS Section 3.6.2.1.c LCO requires suppression pool temperature $\leq$ 110°F when thermal power is $\leq$ 1% RTP) and minimum ECCS. This analysis shows a peak long term suppression pool temperature of about 140°F, which represents a 45°F margin between the calculated pool temperature and the design temperature for the suppression pool (185°F). For the cavity drain case, the proposed special operations TS limits peak suppression pool temperature to $\leq$ 95°F, thus providing additional margin.

4.3.5 Dose Analysis

The DBA LOCA dose calculation credits the CS mode of the Residual Heat Removal (RHR) system for fission product scrubbing in Containment. The CS is not required and will not automatically start for a large line break LOCA in MODE 3 with vessel pressure less than 235 psig because Containment pressure remains below the 7.5 psig analytical limit for spray actuation. An analysis has been performed using current approved methods to determine the impact on control room and exclusion area boundary radiological dose consequences of a DBA LOCA initiated at the defined MODE 3 conditions (235 psig vessel pressure, 3 hours subcritical) with Containment pressure $\leq$ 7.5 psig. The calculated total offsite and control room doses are bounded by corresponding results from the full power DBA.

4.3.6 Small Break LOCA with Steam Bypass of the Suppression Pool

The concept of the pressure suppression reactor Containment is that any steam released from the primary system will be channeled into the suppression pool through the vent system. The steam will be condensed by the suppression pool and will not have an opportunity to produce a significant pressurization effect on the Containment. If a leakage path were to exist between the Drywell and Containment, the leaking steam would pressurize the Containment. The limiting break size for the steam bypass capability analysis is a small break that will not automatically depressurize the reactor vessel and not clear the top LOCA vents. This case maximizes the period of blowdown flow and mass transfer to the Containment through the Drywell bypass area. When the suppression pool level is increased, the pressure in the Drywell required to clear the top vent is also increased. An analysis was performed using GOTHIC to determine the impact of raising the suppression pool level on this steam bypass capability analysis at reduced vessel pressure of 235 psig. The analysis inputs included a break area of approximately 0.07 ft^2, Drywell bypass leakage of $A/\sqrt{K} = 0.9\ ft^2$, no CS, one loop of suppression pool cooling, and structural heat sinks in Containment. Figure 4.10 of Attachment 4 shows a peak Containment pressure of 23.14 psia (8.44 psig), which is below the Containment design pressure (15 psig). The peak pressure is also lower than the DBA case represented in FSAR Figure 6.2-24 (Reference 8).

The above analysis was run without CS to eliminate concerns of entrapment of spray fluid in the drained portion of the reactor cavity pool that can not return to the suppression pool. However, in a small steam line break LOCA, the Drywell pool does not form or forms only minimally. Since the Drywell pool volume is approximately 2 times the spray hold up volume formed by the reactor cavity when it is fully drained, there is sufficient margin in the available suppression pool volume to accommodate the spray hold-up for this event. For bypass leakage cases, Containment pressure will exceed the automatic spray actuation setpoint (7.5 psig analytical) and sprays will operate. Therefore, peak Containment pressure will actually be considerably less than that shown in Figure 4.10.

4.3.7 Large Break LOCA with Steam Bypass of the Suppression Pool

Additional bypass leakage capability studies were performed in response to Humphrey Issues 5.1 and 9.2 (Reference 7). A sensitivity study of a full spectrum of break sizes from Small Break Accident (SBA) to DBA was conducted to determine Containment pressurization prior to the initiation of CS. This assessment assumed a Drywell bypass leakage of $A/\sqrt{K} = 0.9\ ft^2$, CS at 13 minutes, and allowed for structural heat sinks and reactor vessel level control per EOP's to

minimize overflow through the break. The most limiting break was determined to be a 2.5 ft^2 steam line break. Analysis of this event shows Containment pressure increases to 29.2 psia prior to initiation of CS. An analysis was performed using GOTHIC to determine the impact of raising the suppression pool level on this 2.5 ft^2 steam line break bypass leakage capability analysis at reduced vessel pressure of 235 psig. The Containment pressure response in Figure 4.11 of Attachment 4 shows a peak Containment pressure of 26.93 psia prior to initiation of CS, which is below the 29.2 psia predicted for the full power analysis.

CS actuation while in the proposed Special Operations LCO represents an additional suppression pool entrapment volume in the drained reactor cavity equal to about 30% of the Drywell pool volume. Therefore, for the larger break bypass leakage events, the volume of water trapped in the Drywell pool must be limited to 70% of the total volume to ensure adequate suppression pool inventory. This can be accomplished by operator action to control reactor water level (per Emergency Operating Procedures) to Level 8 to limit liquid spillage from the break into the Drywell pool. The most limiting break size for bypass leakage response time is a MSLB DBA (3.54 ft^2) with maximum ECCS. This case maximizes spillage from the break and the rate of Drywell pool formation. A GOTHIC analysis of the 3.54 ft^2 special bypass leakage capability event was performed assuming that operators must control reactor vessel within 7.5 minutes after accident initiation. Once the reactor vessel level is reduced below the break elevation, continued steaming will slowly deplete the suppression pool inventory as the steam from the break condenses and is entrapped in the Drywell pool. Assuming that two loops of CS start at 10.75 minutes and run continuously and all steam from the break condenses in the drywell pool, external makeup flow to the Containment will be required no earlier than 6 hours after accident initiation. If operators control vessel level within 10 minutes, external makeup flow will be required within 1 hour 23 minutes after accident initiation. Since this event assumes maximum ECCS, all divisional power is available and, considering the low pressure MODE 3 conditions, the 10 minute operator response time is reasonable. The required makeup flow can be provided from a number of redundant and diverse sources, including the Feedwater system, Condensate Storage Tank (CST) via the RCIC or High Pressure Core Spray (HPCS) system, and Standby Service Water (SSW) crosstie.

This special bypass capability analysis was also evaluated using GOTHIC assuming minimum ECCS. For this event, the required operator response time to control reactor vessel level increases to 10 minutes and, assuming continuous operation of one spray loop beginning at 10.75 minutes, the time required for external makeup to Containment is about 9 hours 30 minutes.

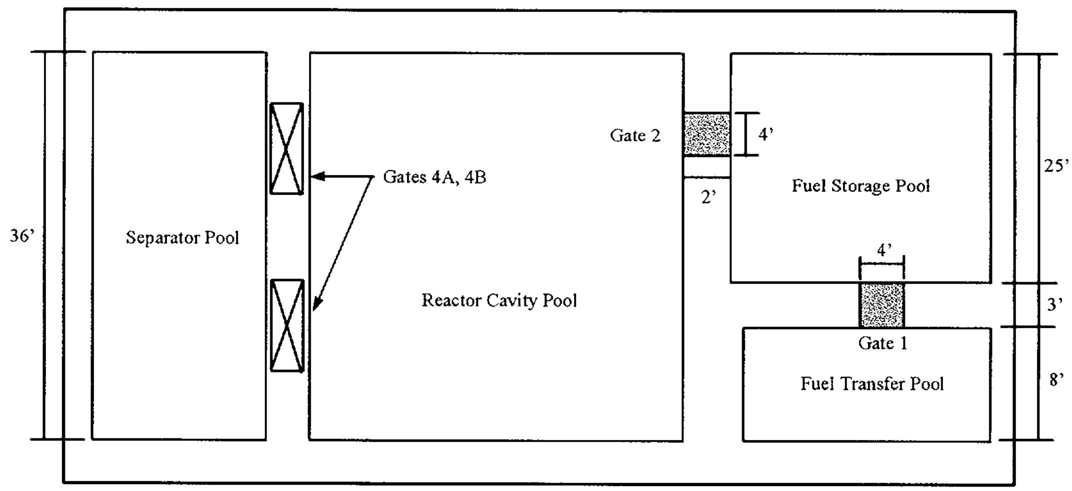
The above events represent beyond design basis bypass leakage capability analyses and are very conservative in that a very large DBA break is postulated to occur coincident with the maximum design bypass leakage and sprays operate continuously. The large break bypass leakage capability analysis with maximum ECCS is the limiting event for operator response time and time to provide external makeup water to the Containment.

4.3.8 Miscellaneous Considerations

In addition to the issues discussed above, there are several less significant items related to suppression pool level. To prevent Drywell flooding during normal operation and transients, the weir wall height is designed to prevent overflow into the Drywell. The normal freeboard from high water level to top of the weir wall is 5 ft 6 inches. An upper pool dump with the maximum amount of water available from the upper pool raises the suppression pool level by about 5 feet. With the suppression pool elevated by 21.25 inches, the freeboard is reduced to about 3 ft 9 inches. However, an upper pool dump at the most limiting point in the cavity drain evolution raises the suppression pool level about 2 ft 2 inches leaving about 1 ft 7 inches of weir wall height available to prevent overflow into the Drywell. The TS Section 3.6.5.4 "Drywell Pressure" limit on Drywell-to-Containment differential pressure of minus 0.26 psid prevents overflow into the Drywell even with the higher starting pool water elevation and an upper pool dump.

The SPMU system design requires that the makeup water addition from the UCP be within an allowable "dump time," defined to be less than or equal to the minimum "pump time." The dump time includes the maximum dump valve opening time. The pump time is determined by dividing the pumping volume by the ECCS pumping rate. The pumping volume considers the suppression pool makeup volume, which is reduced following gate installation and reactor cavity drain. An analysis of the SPMU dump time versus pump time was performed for operations with gates installed and with the reactor cavity pool drained. With gates installed, the allowable dump time is 2 seconds less than the pump time. With the reactor cavity pool drained, the allowable dump time is 5 seconds less than the pump time. Therefore, the SPMU "dump time" criterion is met considering the decreased makeup volume available during the proposed operating conditions.

Figure 1: Upper Containment Pool Arrangement, Water Levels and Elevations (Not to Scale)



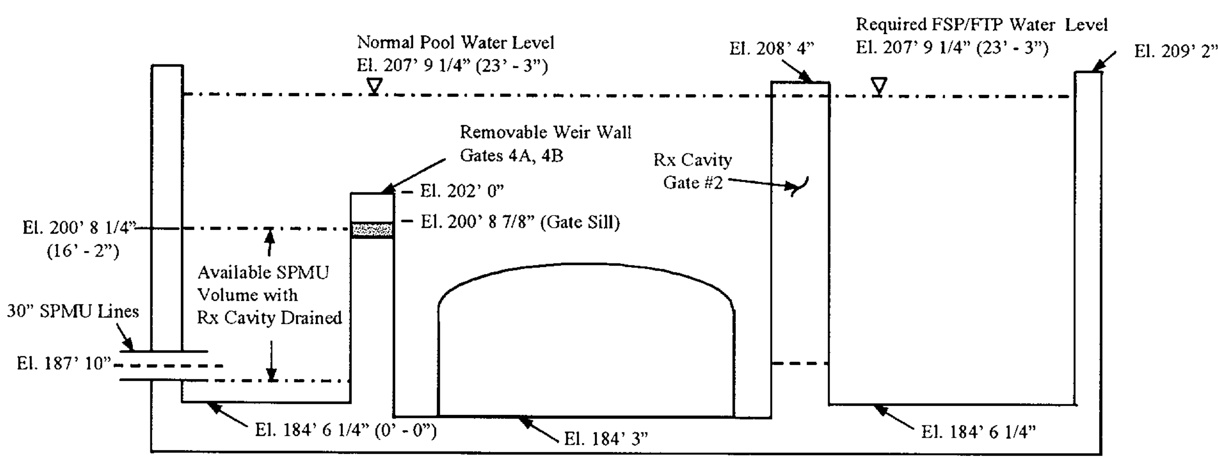
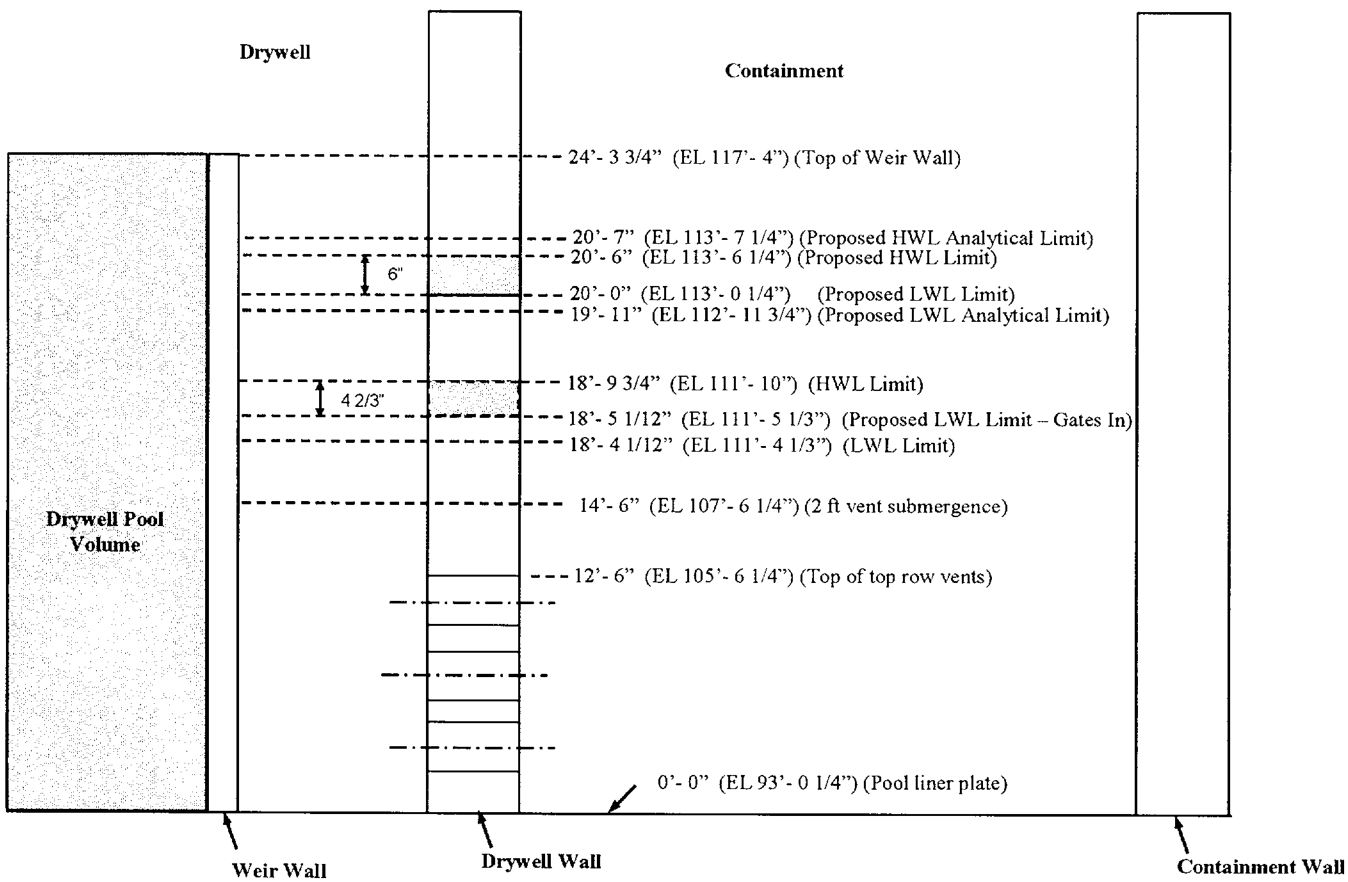


**Figure 2: Suppression Pool Water Levels
(Not to Scale)**



5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. The application provides sufficient information to demonstrate that the request does not alter compliance with any applicable regulatory requirement or criteria.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the FSAR.

5.2 No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Entergy Operations Inc. Energy Company, (Entergy) hereby requests amendment of Facility Operating License for Grand Gulf Nuclear Power Station (GGNS). Specifically, Entergy requests to add a new Special Operations LIMITING CONDITION FOR OPERATION 3.10.9, "Suppression Pool Makeup-MODE 3" and requests modification to the applicability of the Upper Containment Pool (UCP) gates surveillance requirement (TS Section 3.6.2.4, "Suppression Pool Makeup (SPMU) System) to allow installation of UCP gates in MODE 1, 2, or 3. Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes revise the required water levels in the UCP and suppression pool. The probability of an accident previously evaluated is unrelated to the water levels in the pools since they are mitigative systems. The operation or failure of a mitigative system does not contribute to the occurrence of an accident. No active or passive failure mechanisms that could lead to an accident are affected by these proposed changes.

   The consequences of a previously evaluated accident are not significantly increased. The changes have no impact on the ability of any of the Emergency Core Cooling Systems (ECCS) to function adequately, since adequate net positive suction head (NPSH) is provided. The post-accident Containment temperature is not significantly affected by the proposed reduction in total heat sink volume. The increase in suppression pool water level to compensate for the reduction in UCP volume will provide reasonable assurance that the minimum post-accident vent coverage is adequate to assure the pressure suppression function of the suppression pool is accomplished. The suppression pool water level will be raised above the current high water limit for the proposed Special Operations LCO only after the reactor pressure has been reduced sufficiently to assure that the hydrodynamic loads from a loss of coolant accident will not exceed the design values. The reduced reactor pressure will also ensure that the loads due to main steam safety relief valve actuation with an elevated pool level are within the

design loads. The reduced post-LOCA Containment pressure ensures that post-accident dose consequences with no fission product scrubbing by Containment Spray (CS) is bounded by the DBA LOCA .

Therefore, the proposed changes do not significantly increase the consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes to the water level requirements for the UCP and the suppression pool do not involve the use or installation of new equipment. Installed equipment is not operated in a new or different manner. No new or different system interactions are created, and no new processes are introduced. The increased suppression pool water level does not increase the probability of flooding in the Drywell. No new failures have been created by the change in the water level requirements.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes to the UCP and suppression pool water levels do not introduce any new setpoints at which protective or mitigative actions are initiated. No current setpoints are altered by this change. The design and functioning of the Containment pressure suppression system is unchanged. The proposed total water volume is sufficient to provide high confidence that the pressure suppression and Containment systems will be capable of mitigating large and small break accidents. All analyzed transient results remain well within the design values for the structures and equipment. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 <u>Environmental Considerations</u>

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 References

1. FSAR Appendix 6A, "Grand Gulf Containment Loads."
2. FSAR Appendix 6D, "Mark III Containment Loads" (Appendix 3B to GESSAR II).
3. FSAR Table 6.2-5, "Summary of Short Term Accident Results for Containment Response to Recirculation Line and Steamline Breaks."
4. NUREG-0978, Mark III LOCA-Related Hydrodynamic Load Definition, August 1984
5. General Electric Report NEDE-24648-P, "MARK III Confirmatory Test Program 1/9-Area Scale Multivent Pool Swell Tests," August 29, 1979.
6. General Electric Design Report 22A4365, "Interim Containment Loads Report Mark III Containment," Revision 1, April 1978.
7. Letter from L. L. Kintner (U.S. NRC) to O. D. Kingsley (SERI), "SER Relating to Humphrey Concerns," dated March 23, 1987.
8. FSAR Figure 6.2-24, "Containment Pressure Versus Time" [Following a Small Break with Steam Bypass with Containment Spray and Heat Sinks].
9. Letter from L. F. Dale (MP&L) to H. R. Denton (USNRC), "Humphrey Containment Concerns," dated March 23, 1983 (AECM-83/0146).

ATTACHMENT 2 TO GNRO-2002/00011

PROPOSED TECHNICAL SPECIFICATION CHANGES

3.10 SPECIAL OPERATIONS

3.10.9 Suppression Pool Makeup-MODE 3

LCO 3.10.9 The requirements of LCO 3.6.2.2, "Suppression Pool Water Level" and LCO 3.6.2.4, "Suppression Pool Makeup (SPMU) System," may be suspended in MODE 3 to allow drain-down of the Upper Containment Pool, provided the following requirements are met:

a. Suppression Pool Average Temperature is $\leq$ 95°F;

b. Suppression Pool and Upper Containment Pool water levels are maintained within limits of Figure 3.10.9-1;

c. The fuel storage and transfer canal areas of the Upper Containment Pool are maintained at a minimum of 23 ft 3 inches.

d. Reactor Steam Dome pressure is < 230 PSIG;

e. Reactor has been subcritical > 3 hours; and

f. Each SPMU subsystem valve is OPERABLE per SR 3.6.2.4.3 and SR 3.6.2.4.5 and Upper Containment Pool temperature is in compliance with SR 3.6.2.4.2.

APPLICABILITY: MODE 3 with LCO 3.6.2.2 and 3.6.2.4 not met.

ACTIONS

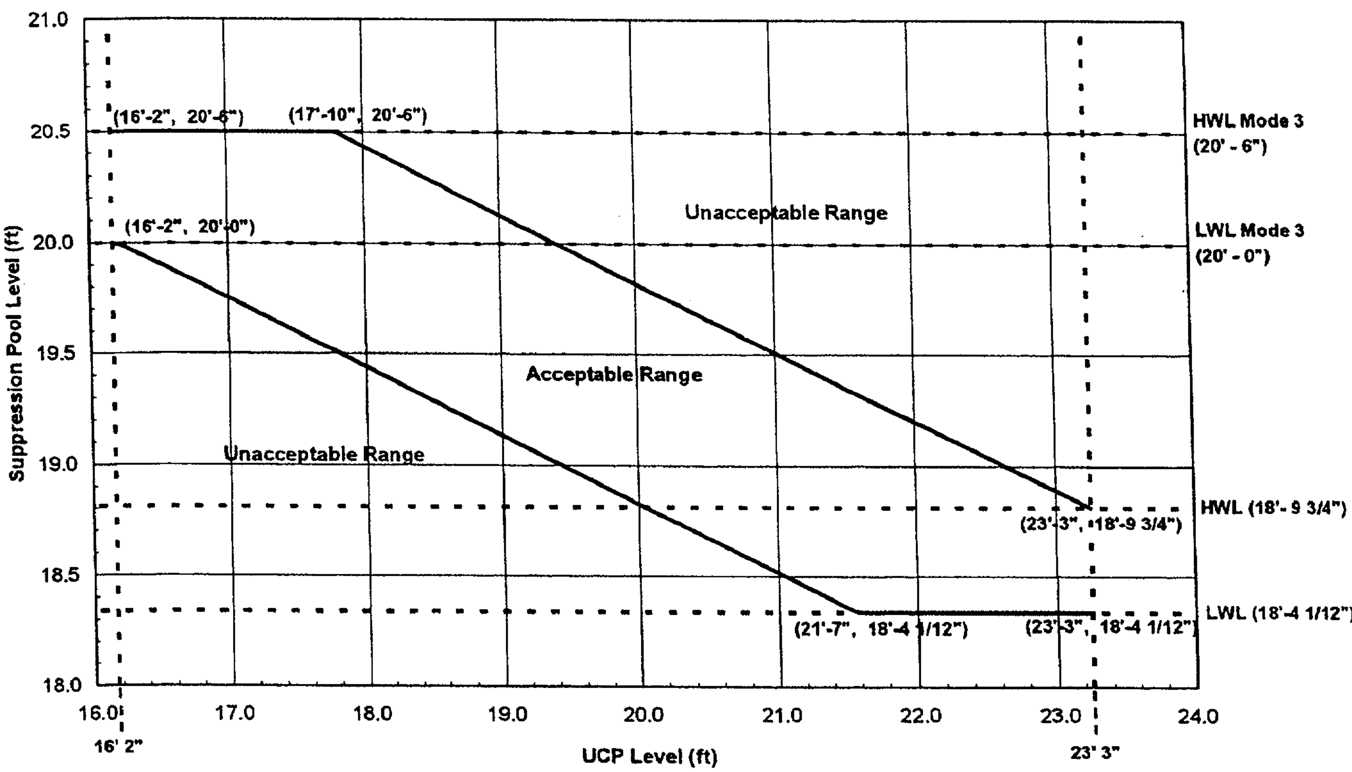
------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each requirement of the LCO.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more of the above requirements not met.	A.1	Suspend draining the Upper Containment Pools.	Immediately
	<u>AND</u>		
	A.2	Restore compliance with LCO requirements.	4 hours
B. Required Actions and Completion Time of condition A not met.	B.1	Restore compliance with suspended MODE 3 requirements.	12 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 4.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.9.1	Verify Suppression Pool temperature ≤ 95°F.	12 hours
SR 3.10.9.2	Verify Reactor Steam Dome pressure is < 230 PSIG.	12 hours
SR 3.10.9.3	Verify level in the Upper Containment Pool and the Suppression Pool to be within limits of Figure 3.10.9-1.	12 hours
SR 3.10.9.4	Verify level in the fuel storage and transfer canal areas of the Upper Containment Pool are ≥ of 23 ft 3 inches.	12 hours

Figure 3.10.9-1
Upper Containment and Suppression Pool Levels



SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.4.1	Verify upper containment pool water level is ≥ 23 ft 3 inches above the pool bottom.	24 hours
SR 3.6.2.4.2	Verify upper containment pool water temperature is ≤ 125°F.	24 hours
SR 3.6.2.4.3	Verify each SPMU subsystem manual, power operated, and automatic valve that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.2.4.4	-----------------NOTE------------------ The requirements of this SR are not required to be met when all upper containment pool levels are maintained per SR 3.6.2.4.1 and suppression pool water level is maintained ≥ 18 ft 5 1/12 inches (one inch above LCO 3.6.2.2 Low Water Level). -- Verify all upper containment pool gates are in the stored position or are otherwise removed from the upper containment pool.	31 days
SR 3.6.2.4.5	-----------------NOTE------------------ Actual makeup to the suppression pool may be excluded. -- Verify each SPMU subsystem automatic valve actuates to the correct position on an actual or simulated automatic initiation signal.	18 months

ATTACHMENT 3 TO GNRO-2002/00011

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

B 3.10 SPECIAL OPERATIONS

B 3.10.9 Suppression Pool Makeup System

BASES

BACKGROUND

Maintaining the SPMU inventory in the Upper Containment Pools will lead to delays in completing outage work in a timely manner, particularly with the advent of things like Noble Metal addition technology has led to the need for holding temperature and pressure at a point above the MODE 4 definition of Table 1.1-1.

The purpose of this Special Operations LCO is to allow the Upper Containment Pool to be drained below its normal level such that certain refueling activities can proceed prior to reaching MODE 4. These activities include installation of the gate between the refueling cavity and the upper containment (fuel storage pool) and completely draining the reactor cavity.

APPLICABLE SAFETY ANALYSES

Supporting analyses and engineering calculations determined the required water inventory to ensure that the suppression pool makeup function is satisfied if the specified conditions of this Special Operations LCO are met. Supporting analyses differ from those for TS 3.6.2.4 in that a portion of the SPMU volume is assumed to have already been transferred to the suppression pool with the remainder available from the separator storage pool portion of the Upper Containment Pool. These analyses demonstrate that the containment spray function of RHR is not required following a design basis LOCA to protect the containment given the reduced temperature and pressure stipulated by the LCO. An empty reactor cavity creates a large hold-up volume that would significantly deplete the suppression pool inventory if containment spray operation were to occur. The analysis results demonstrate that the containment pressure increase following a DBA LOCA will not be sufficient to result in the auto-initiation of containment spray.

In addition to the design basis analyses, drywell bypass capability analyses (Reference 1) indicate that containment pressure could exceed the containment spray auto-actuation setpoint. Steam bypass leakage and the associated capability analyses are discussed in Reference 4. For the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

most limiting large break bypass leakage capability analysis (Ref. 1), operator action to control reactor water level is credited in ensuring that sufficient inventory is available for containment spray operation.

The containment loads evaluation performed for this special operation including the elevated suppression pool levels demonstrates that at the decay time and reactor pressure specified by the LCO, the containment loads are bounded by those calculated for the DBA LOCA.

Specific analyses demonstrate containment temperature and pressure as well as radiological consequences are bounded by those following large and small break LOCAs at full power conditions. The applicable analyses supporting the LCO are contained in References 1, 2 and 3. During these events, the SPMU System is relied upon to dump the separator pool water to maintain drywell horizontal vent coverage and an adequate suppression pool heat sink volume to ensure that the primary containment internal pressure and temperature stay within design limits.

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Operation with the Upper Containment Pool levels below those specified in SR 3.6.2.4.1 can be achieved by exiting the condition where LCO 3.6.2.4 applies. Operation with elevated suppression pool levels is also optional as operation at levels above those specified in LCO 3.6.2.2 can be achieved by exiting the condition where the LCO applies.

Compliance with the Figure 3.10.9-1 level requirements ensure that there is sufficient overlap with the requirements of LCO 3.6.2.2 and 3.6.2.4 such that the volume in containment during the transition to a drained refueling

(continued)

BASES

LCO (continued)	cavity fulfills the containment water inventory requirements assumed in the analysis. Once the level of the weir wall separating the refueling cavity from the separator storage pool is reached, Figure 3.10.9-1 only applies to the separator pool. Supporting analyses assume that the weir wall gates are not installed. Maintaining the fuel storage and transfer canal area pools ensures that water traps inside containment are minimized consistent with the supporting analysis. The reactor subcritical time, suppression pool average temperature, and reactor steam dome pressure are assumptions of the supporting analyses. Entry into MODE 4 operation does not require the use of this Special Operations LCO or its ACTIONS.
APPLICABILITY	The MODE 3 requirements may only be modified for allowing early drain-down of the Upper Containment Pool while performing Noble Metal addition or during a reactor cool down for a refueling outage. The requirements of this LCO provide conservatism in the response of the unit to any event that may occur. Operations in all other MODES are unaffected by this LCO.
ACTIONS	A Note has been provided to modify the ACTIONS related to drain-down of Upper Containment Pools-MODE 3. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for each requirement of the LCO not met provide appropriate compensatory measures for separate requirements that are not met. As such, a Note has been provided that allows separate entry for each requirement of the LCO.

(continued)

BASES

ACTIONS (continued)

A.1

With the requirements of the LCO not met (e.g., Upper Containment Pool level not within limits), the draining of the Upper Containment Pool is to be suspended. Thereby, a worsening of the circumstances will be prevented.

A.2

If one or more of the requirements of this Special Operations LCO are not met, the immediate implementation of the Required Action commences activities, which will restore operation consistent with the Special Operations LCO. The Completion Time is intended to require that these Required Actions be implemented in a very short time and carried through in an expeditious manner.

B.1

Required Action A.2.2 is an alternative Required Action that can be taken instead of Required Action A.2.1 to restore compliance with the normal MODE 3 requirements, thereby exiting this Special Operations LCOs Applicability. The allowed Completion Time allows sufficient time to reestablish compliance with the appropriate Technical Specification.

B.2

If the requirements of this Special Operations LCO or the normal MODE 3 requirements cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions and is consistent with the time provided in LCO 3.0.3 for reaching MODE 4 from MODE 3.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.10.9.1 and SR 3.10.9.2

Verification of the Suppression Pool temperature and Steam Dome pressure ensures that assumptions of the supporting analyses for this Special Operations LCO are continually met. Therefore, the plant response to an accident while in this Special Operations LCO will remain bounded by the Design Basis Loss of Coolant Accident.

The Frequency of 12 hours is based on engineering judgement and is considered adequate due to the unlikely event of unknowingly adding heat to the Suppression Pool or increasing Reactor pressure.

SR 3.10.9.3

Verification of the required Upper Containment Pool and Suppression Pool levels to be within limits ensures that the engineering assumptions for the calculations supporting this Special Operations LCO are continually met. These assumptions ensure sufficient inventory is available such that Drywell vent submergence and Suppression Pool heat sink requirements are met.

The Frequency of 12 hours is based on engineering judgement and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

SR 3.10.9.4

Verification of the required Fuel Storage and Transfer Canal Pool levels to be within limits ensures that the engineering assumptions for the calculations supporting this Special Operations LCO are continually met. These assumptions ensure sufficient inventory is available such that Drywell vent submergence and Suppression Pool heat sink requirements are met.

The Frequency of 12 hours is based on engineering judgement and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

(continued)

BASES (continued)

REFERENCES

1. Calculation XC-Q1M10-01012, "MODE 3 Containment Analysis at Reduced Reactor Pressure."
2. Calculation XC-Q1E30-01004, "Suppression Pool Makeup System - MODE 3."
3. Calculation XC-Q1111-01011, "MODE 3 LOCA Dose Analysis."
4. UFSAR 6.2.1.1.5.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.4.2

The upper containment pool water temperature is regularly monitored to ensure that the required limit is satisfied. The 24 hour Frequency was developed based on operating experience related to upper containment pool temperature variations during the applicable MODES.

SR 3.6.2.4.3

Verifying the correct alignment for manual, power operated, and automatic valves in the SPMU System flow path provides assurance that the proper flow paths will exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control and because improper valve position would affect only a single subsystem. This Frequency has been shown to be acceptable through operating experience.

SR 3.6.2.4.4

The upper containment pool has two gates used to separate the pool into distinct sections to facilitate fuel transfer and maintenance during refueling operations and two additional gates in the separator pool weir wall extension, which, when installed, limit personnel exposure and ensure adequate water submergence of the separator when the separator is stored in the pool. The SPMU System dump line penetrations are located in the steam separator storage section of the pool. To provide the required SPMU System dump volume to the suppression pool, the gates must be removed (or placed in their stored position) to allow communication between the various pool sections. The Surveillance is modified by a Note that allows leaving

(continued)

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.

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.

ATTACHMENT 4 TO GNRO-2002/00011

INFORMATION RELATED TO THE GOTHIC COMPUTER CODE

The proposed revision to the GGNS Technical Specifications to add a new Special Operations LCO allows draining of the reactor cavity pool portion of the UCP during MODE 3, "Hot Shutdown," with reduced pressure in the reactor pressure vessel (RPV). Since a portion of the water in the reactor cavity pool is used for makeup to the suppression pool, the suppression pool minimum water level is increased to compensate for the decreased inventory.

The analyses supporting the proposed change were performed using the GOTHIC (Generation of Thermal-Hydraulic Information for Containments) computer program. GOTHIC is an advanced computer program used to perform transient thermal hydraulic analyses of multiphase systems in complex geometries. GOTHIC solves the conservation equations for mass, momentum, and energy for multicomponent, multiphase flow. As documented in the GOTHIC Qualification Report (Reference 4.1), GOTHIC predicted solutions have been compared to analytical solutions and to experimental data for Containment applications. GOTHIC has been previously used for Containment, high energy line break, and heating and ventilation analyses at numerous nuclear power plants.

Examples of GOTHIC Containment analysis applications previously reviewed by the Commission include:

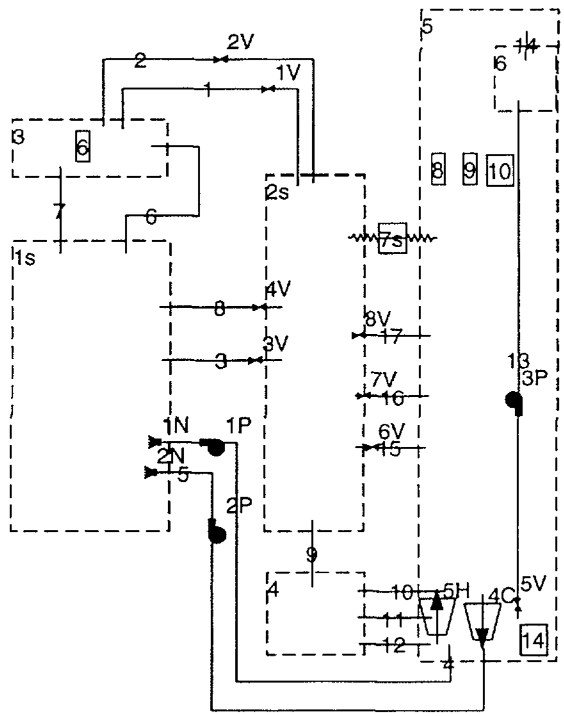
- The adoption of the GOTHIC code for Containment analysis by Kewaunee Nuclear Power Plant (Reference 4.2)
- Evaluation of Containment Cooling by Waterford Steam Electric Station, Unit 3 (Reference 4.3)
- Evaluation of the AP600 design with WGOTHIC (Reference 4.4).

At GGNS, plant-specific benchmarking was performed to the current licensing basis Containment analyses in Chapter 6.2 and Appendix 15D of the Final Safety Analysis Report (FSAR). The benchmarking showed generally conservative correlation of the GOTHIC output data relative to the current FSAR data generated by General Electric (GE) codes.

The GOTHIC models use a nodal diagram similar to the representative nodal diagram shown in Figure 4.1. This diagram is for the long-term, low pressure (MODE 3) main steam line break analysis. Equivalent nodalization is used for the bypass leakage capability analyses.

Values of key input parameters used in the GOTHIC models are provided in the tables in this attachment.

Figure 4.1: Representative Nodalization Diagram for GGNS GOTHIC MSLB Models



Volumes: **1s – Reactor Vessel**
3 - Reactor Vessel Dome
2s - Drywell
4 – Drywell Weir Annulus
5 - Containment
6 – Upper Containment Pool

Initial Conditions and Input Parameters Used in the Analysis

Initial Conditions

Reactor Vessel

The initial conditions for the benchmark analyses were developed from data in FSAR Table 6.2-4 (Reference 4.5) and other GGNS references. The inputs are summarized in the table below.

Parameter	Value Used in the GOTHIC Analysis
Reactor Power Level (MWt)	3995 MWt (104.2% of Rated)
Average Coolant Pressure (psia)	1060
Average Coolant Temperature (°F)	551.7
Volume of Liquid in RPV in Model (ft^3)	13,036
Volume of Steam in RPV in Model (ft^3)	9,564
Total Reactor Coolant Volume in Model (ft^3)	23,809

The total reactor coolant volume includes the volumes of liquid in the reactor recirculation system piping and miscellaneous connected lines and the volume of steam in the four main steam lines to the first MSIV.

Containment

The initial conditions and physical parameters for the current license basis Containment analysis are listed in FSAR Tables 6.2-4 and 6.2-1 (Reference 4.5 and 4.6). The following table provides the values used in the GOTHIC analyses. The values are generally consistent with the FSAR values for the benchmark cases. Minor differences between GOTHIC model nodalization and the original licensing basis methods/codes used for various events may require that inputs be varied from these values for specific cases.

Parameter	Value
Drywell Pressure, psig	0.0
Containment Pressure, psig	0.0
Drywell Air temperature , °F	135
Containment Air temperature , °F	95
Drywell Relative humidity, %	40
Containment Relative humidity, %	60
Suppression pool water, temperature °F (See Note 1)	95 110
Upper pool water temperature, °F	125
Suppression pool depth, ft (Benchmark Cases) (See Note 2)	Low water level: 18' – 4 1/12" High water level: 18' – 9 ¾"
Suppression Pool Water Volume, ft ^{3} (Benchmark Cases) (See Note 2)	Low water level: 132,556 High water level: 136,014
Suppression pool depth, ft (MODE 3 Cases) (See Note 3)	Low water level: 19' – 11" High water level: 20' – 7"
Suppression Pool Water Volume, ft ^{3} (MODE 3 Cases) (See Note 3)	Low water level: 143,768 High water level: 148,552
Suppression Pool Surface Area, ft ^{2}	7220
Upper pool makeup volume, ft ^{3} (See Note 4)	12,333.5
Drywell Free Volume, ft ^{3}	270,000
Containment Free Volume, ft ^{3}	1,400,000

Notes

1 FSAR benchmark cases are performed with an initial pool temperature of 95 °F. The MODE 3 cases at reduced vessel pressure to support the proposed amendment were run at 110°F initial suppression pool temperature.

2 The suppression pool high water level is used for the short-term MSLB benchmark analysis (i.e., first 1800 seconds) evaluating peak Drywell pressure. The suppression pool low water level is used for the longer-term MSLB benchmark analysis evaluating peak suppression pool temperature and Containment pressure. The GOTHIC pool volumes conservatively neglect the volume in the Drywell vents.

3 For the low pressure MODE 3 cases supporting the proposed amendment, the current suppression pool high water level plus 1 ft 9 ¼ inches (20 ft 7 inches) is used for the short-term analysis evaluating peak Drywell pressure. The longer-term analysis uses the current high water level plus 1 ft 1 ¼ inches (19 ft 11 inches). Additional volume from the UCP (i.e., from the suppression pool makeup system) volume is credited for the low pressure MODE 3 analysis.

4 This is the value used in the analyses for the cases with the reactor cavity drained. Benchmark cases crediting upper pool dump use the design value, 36,163 ft^3.

Decay Heat

The decay heat used in the benchmark analyses include the fuel relaxation energy and fission product decay heat based on May-Witt values. The decay heat function is consistent with the current FSAR normalized decay heat in FSAR Table 6.2-12 (Reference 4.7) multiplied by the nominal heat rate of 3995 MWt. In addition to the decay heat, metal water reaction energy equal to 288 BTU/sec is included over the first 1800 seconds. The low pressure MODE 3 analyses starting 3 hours after shutdown use a more realistic (less conservative) but bounding decay heat curve based on ORIGIN calculations.

Available Containment Heat Sinks

The available Containment heat sinks used in the benchmark cases are unchanged from current FSAR values contained in FSAR Table 6.2-9 (Reference 4.8). Realistic conductors based on plant-specific calculations are used in the low pressure MODE 3 cases.

Benchmark Results

Key results of the GOTHIC benchmarks against GE results reported in the FSAR are presented in Figures 4.2 through 4.6. Figures 4.2 and 4.3 show the short-term Drywell pressure and temperature for the main steam line break. These figures show good agreement between GOTHIC and the FSAR values. The GOTHIC peak Drywell pressure is 22.4 psig, which compares to 22 psig in the FSAR. The lower GOTHIC values of Drywell pressure and temperature observed during the period following the pressure peak is due to the cooling effect of liquid droplets from the break. This effect is not included in the original GE methodology. Figures 4.4 and 4.5 show the long-term suppression pool temperature and Containment pressure after the main steam line break with minimum ECCS. These results also show good agreement with the FSAR values. The GOTHIC peak suppression pool temperature is about 8°F higher than the FSAR value. The GOTHIC peak Containment pressure is 24.3 psia which is slightly below the FSAR peak value of 24.6 psia. Figure 4.6 shows the Containment pressure resulting from suppression pool bypass leakage ($A/\sqrt{K}$ equal to 0.9 square foot) following a small break accident. This figure shows excellent agreement between GOTHIC and the FSAR Containment pressure peak. The GOTHIC peak pressure is 29.76 psia, which is slightly higher (and conservative) relative to the GE/FSAR value.

MODE 3 Results with Reactor Vessel Pressure Starting at 235 psig

Key results from the GOTHIC cases supporting the proposed Technical Specification change are shown in Figures 4.7 through 4.11. Figure 4.7 shows the short-term Drywell pressure for a main steam line break with the reactor vessel starting at 235 psig, the suppression pool level starting at the proposed high water level limit (1 foot 9 ¼ inches above current high water level), suppression pool temperature starting at 110°F, and suppression pool makeup via upper pool dump. The peak Drywell pressure is only 25.46 psia, which compares to 37.1 psia from the DBA MSLB benchmark. Figures 4.8 and 4.9 show the long-term suppression pool temperature and Containment pressure for this case with the suppression pool level starting at the proposed low water level (1 foot 1 ¼ inches above current high water level). The peak long-term Containment pressure is 5.94 psig, which is below the lowest Containment Spray (CS) actuation pressure (7.5 psig analytical). The peak suppression pool temperature is 140°F. Figure 4.10 shows the Containment pressure resulting from a small break with suppression pool bypass leakage ($A/\sqrt{K}$ equal to 0.9 square foot), reactor vessel pressure starting at 235 psig, suppression pool temperature starting at 110°F, suppression pool level starting at the proposed high water level limit, and no Containment sprays (or liquid flooding out of the break). The peak Containment pressure is 8.44 psig, slightly above the spray actuation setpoint but well below the Containment design pressure limit (15 psig). Figure 4.11 shows the Containment pressure resulting from the "Humphrey Issues" 2.5 square foot bypass leakage ($A/\sqrt{K}$ equal to 0.9 square foot) analysis with reactor vessel pressure starting at 235 psig, suppression pool temperature starting at 110°F, suppression pool level starting at the proposed high water level limit, and CS after 13 minutes (780 seconds). The Containment pressure increases to about 27 psia at the time of spray initiation.

Figures 4.12 and 4.13 show the integrated break flow (equal to inventory trapped in the Drywell pool) for the 3.54 square foot main steam line break bypass leakage ($A/\sqrt{K}$ equal to 0.9 square foot) studies with reactor vessel pressure starting at 235 psig, suppression pool temperature starting at 110°F, and suppression pool level starting at the proposed high water level limit. Figure 4.12 shows the result from the maximum ECCS case. In this case, ECCS flow is throttled and reactor vessel level lowered to Level 8 (per EOP's) at 7.5 minutes. The inventory trapped in the Drywell pool at 7.5 minutes is equal to about 49% of the total Drywell pool volume. Figure 4.13 shows the integrated break flow from the minimum ECCS case. ECCS flow is throttled and level control is obtained at 10 minutes in this case.

References

4.1. NAI 8907-09, Revision 5, "GOTHIC Containment Analysis Package Qualification Report," Version 6.1, July 1999.

4.2. Letter from J. G. Lamb (US NRC) to M. Reddemann (Nuclear Management Company, LLC), Kewaunee Nuclear Power Plant, "Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3," dated September 10, 2001.

4.3. Letter from N. Kalyanam (US NRC) to C.M Dugger (Entergy Operations, Inc.), "Waterford Steam Electric Station Unit 3, Issuance of Amendment 165, RE:Reduction in Operable Containment Fan Coolers in the Containment Cooling System (TAC No. MA6997)," July 6, 2000.

4.4. NUREG-1512, Safety Evaluation by the NRC for the Westinghouse AP-600 Design

4.5. FSAR Table 6.2-4, Initial Conditions Employed in Containment Response Analyses.

4.6. FSAR Table 6.2-1, Containment Design Parameters.

4.7. FSAR Table 6.2-12, Core Decay Heat Following LOCA for Containment Analyses.

4.8. FSAR Table 6.2-9, Available Heat Sinks.

Figure 4.2: Short Term Main Steam Line Break Benchmark – Drywell Pressure

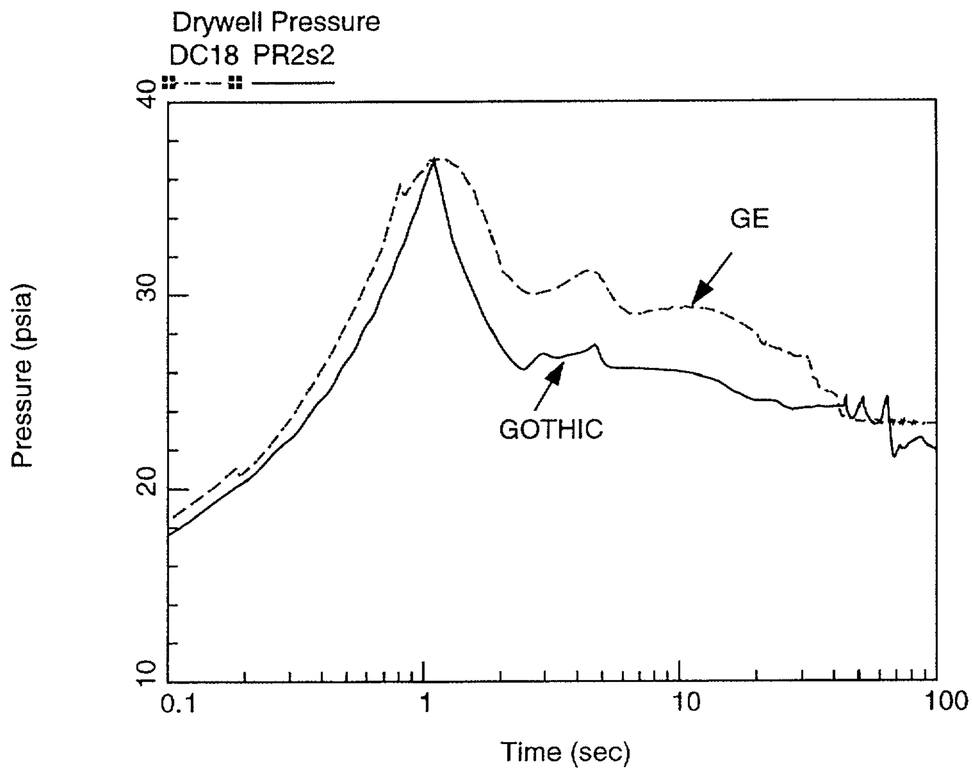


Figure 4.3: Short Term Main Steam Line Break Benchmark – Drywell Temperature

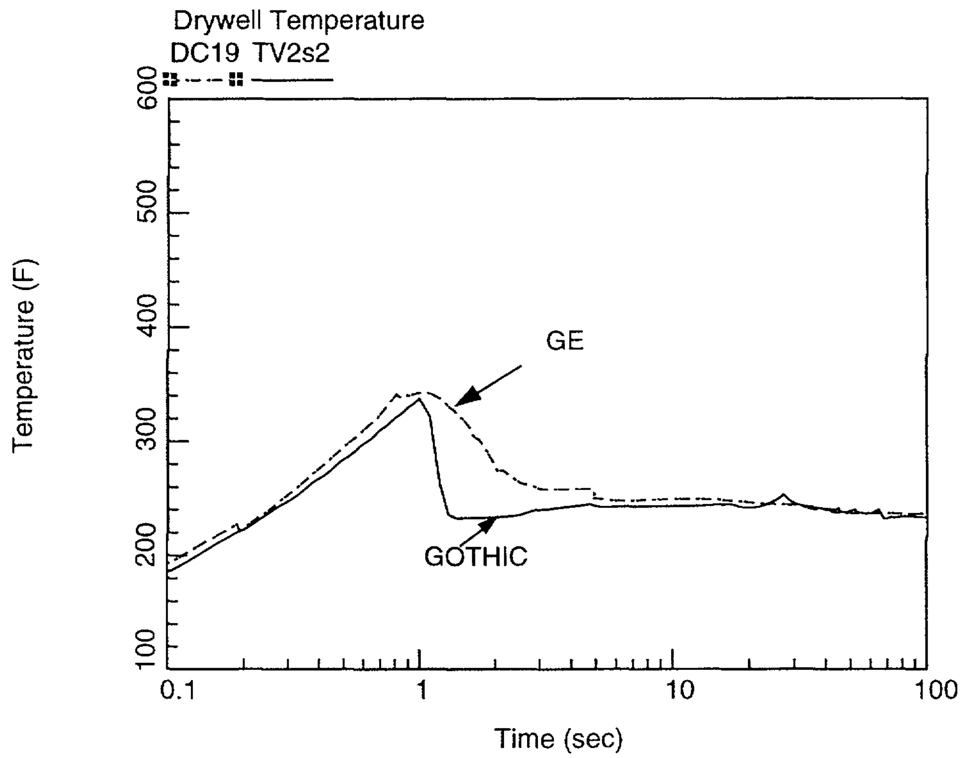


Figure 4.4: Long Term Main Steam Line Break Benchmark – Suppression Pool Temperature

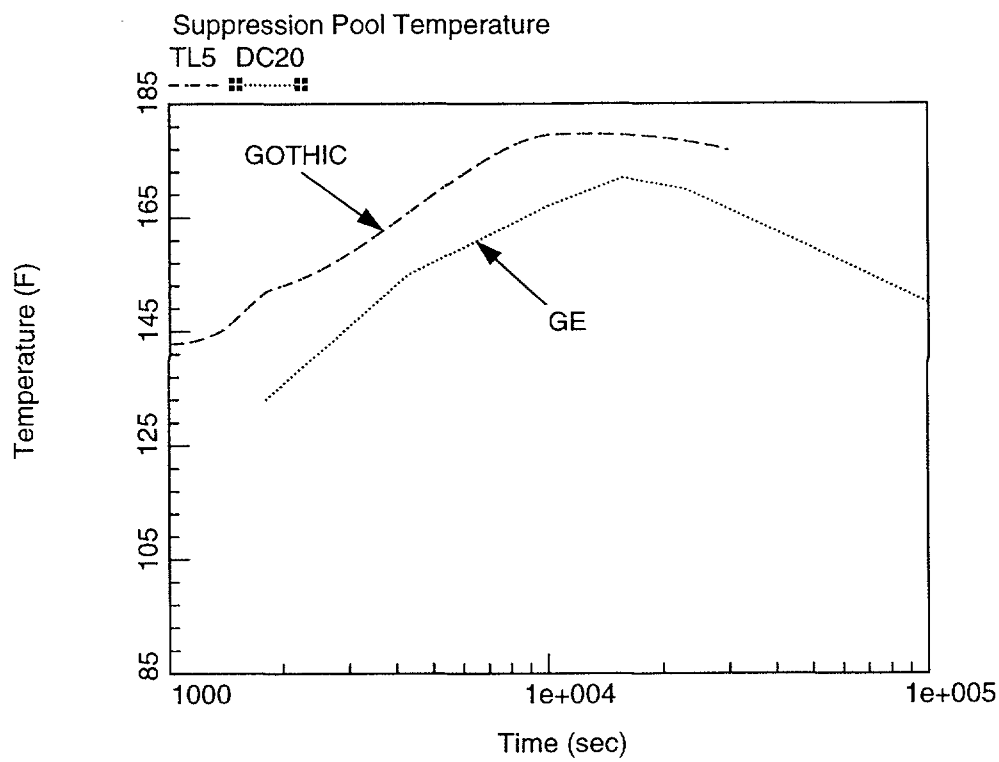


Figure 4.5: Long Term Main Steam Line Break Benchmark – Containment Pressure

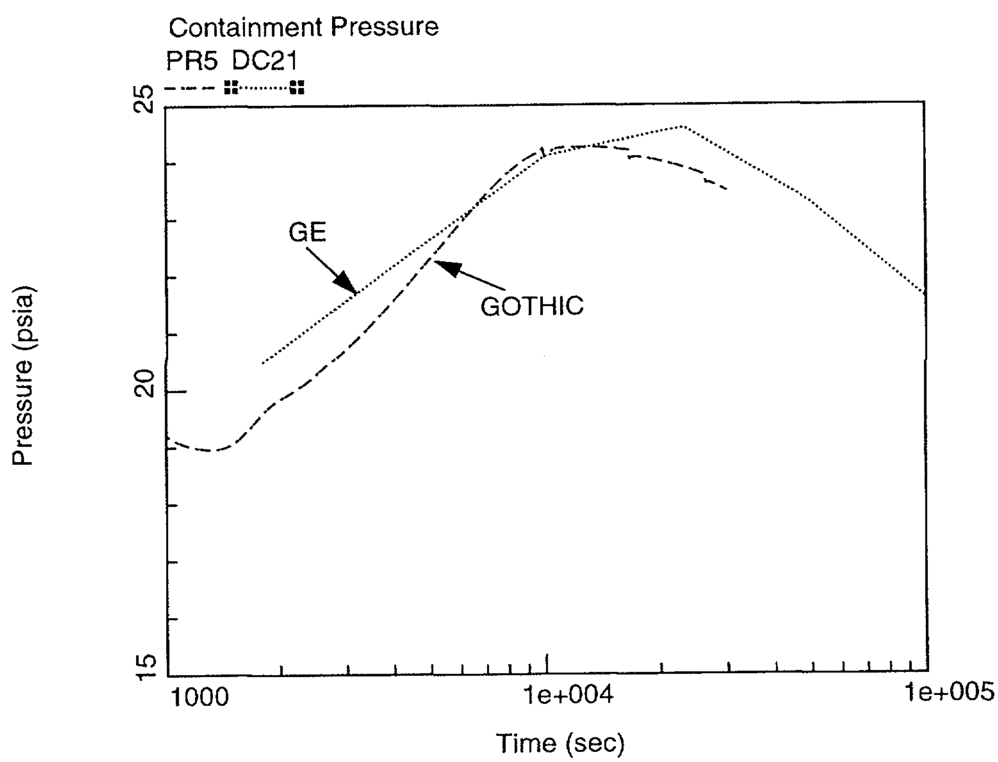


Figure 4.6: Small Break Bypass Leakage Benchmark – Containment Pressure

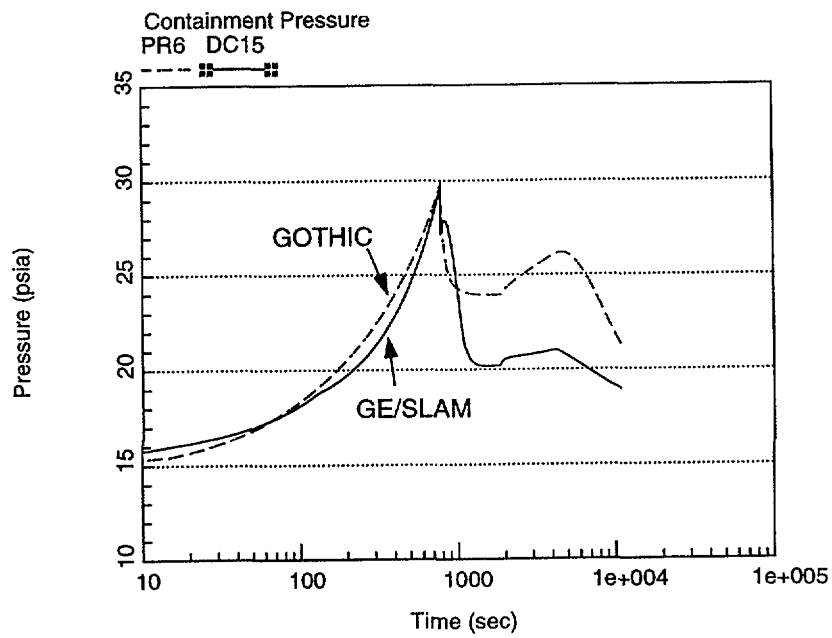


Figure 4.7: Short Term Main Steam Line Break (235 psig RPV Pressure) – Drywell Pressure

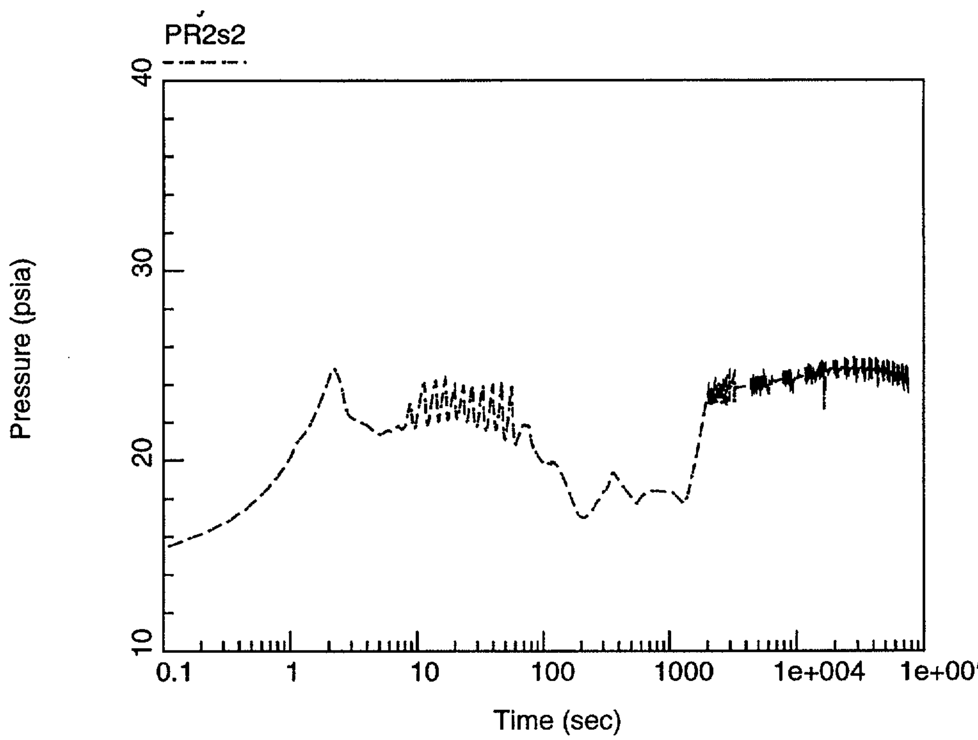


Figure 4.8: Long Term Main Steam Line Break (235 psig RPV Pressure) – Suppression Pool Temperature

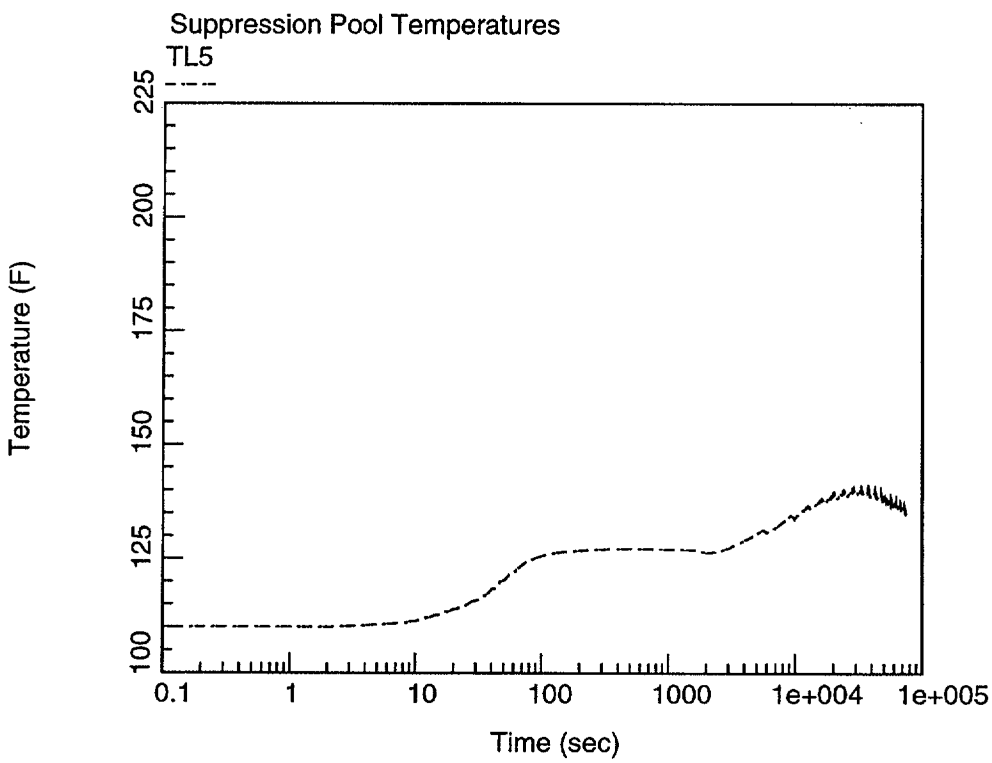


Figure 4.9: Long Term Main Steam Line Break (235 psig RPV Pressure) – Containment Pressure

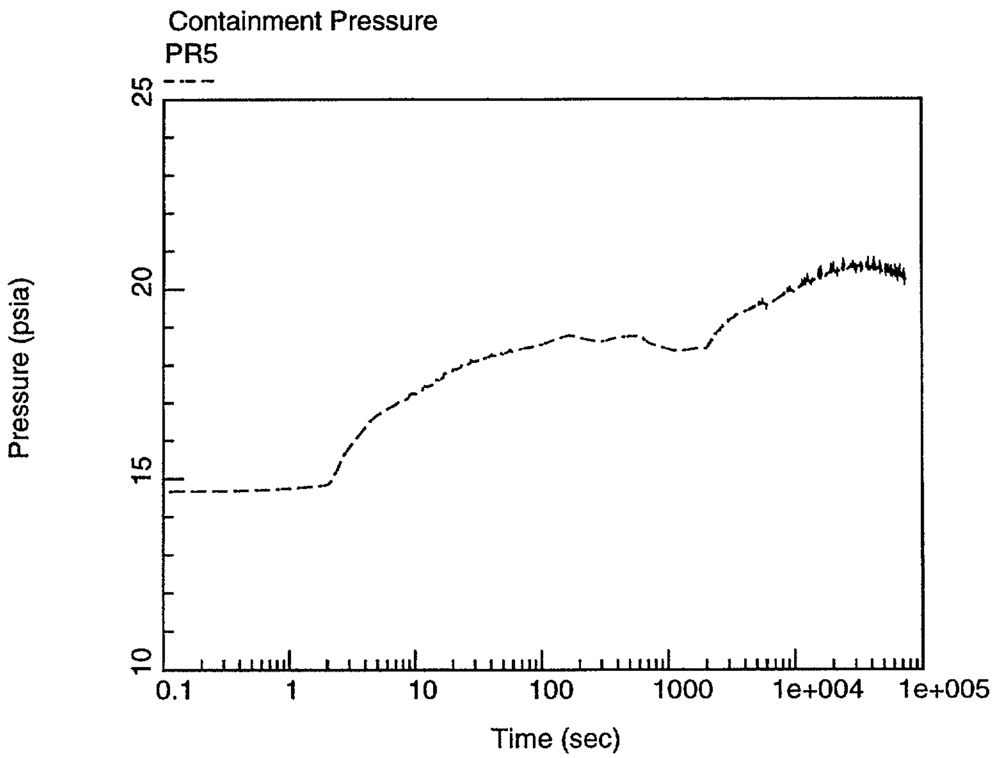


Figure 4.10: Small Break Bypass Leakage (235 psig RPV Pressure) – Containment Pressure

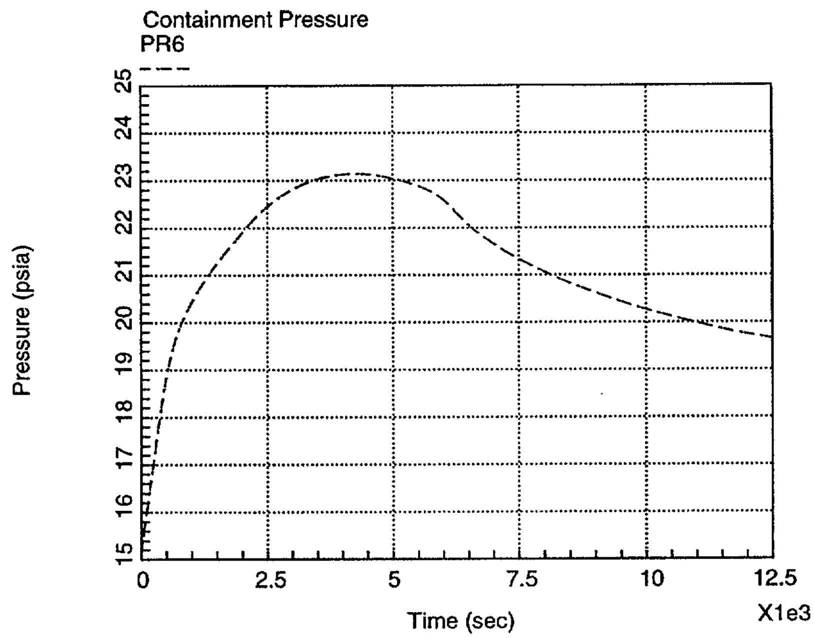


Figure 4.11: 2.5 ft^2 Bypass Leakage (235 psig RPV Pressure) – Containment Pressure

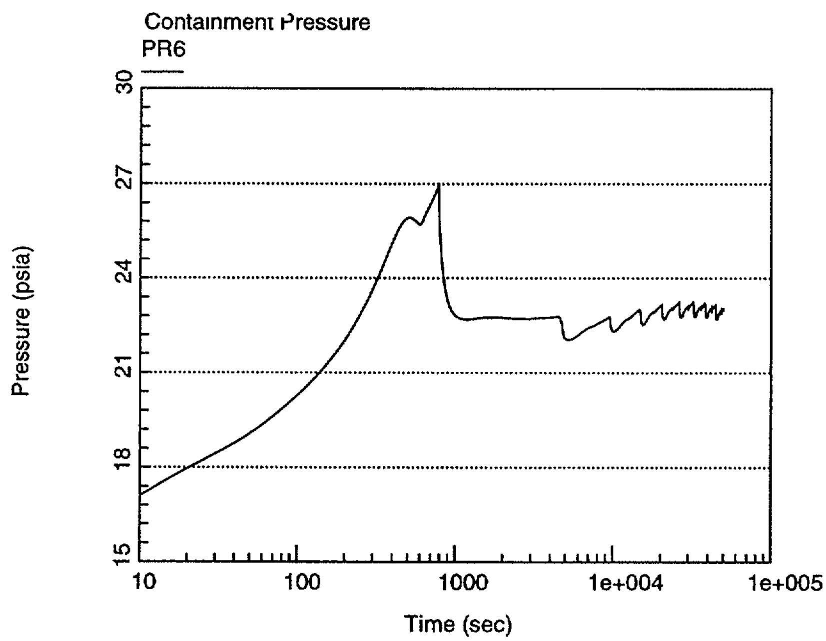


Figure 4.12: 3.54 ft^2 Bypass Leakage (235 psig RPV Pressure) Inventory Loss – Maximum ECCS

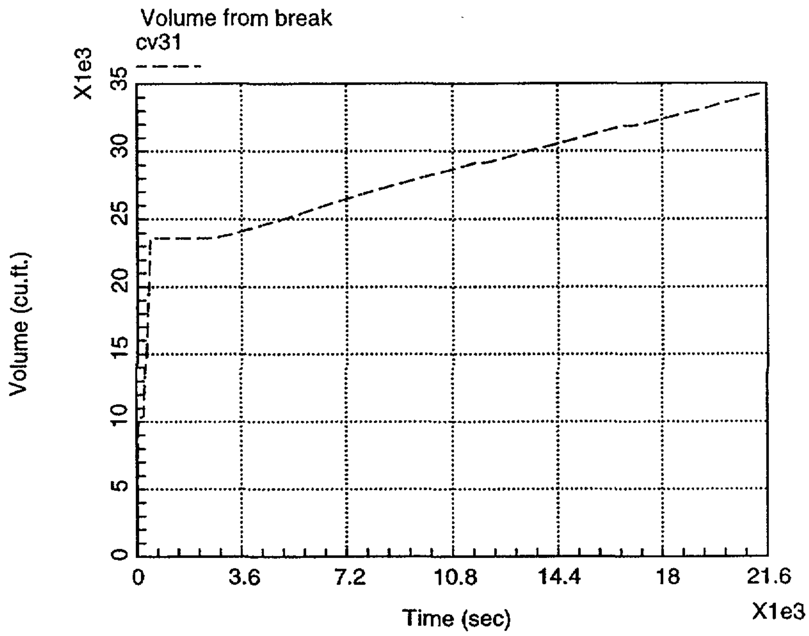


Figure 4.13: 3.54 ft^2 Bypass Leakage (235 psig RPV Pressure) Inventory Loss – Minimum ECCS