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March 15, 2016
L-16-083

10 CFR 50.90

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
License Amendment Request for Upper Containment Pool (UCP) Gate Installation in MODEs 1, 2, and 3, and Drain-Down of the Reactor Cavity Portion of the UCP in MODE 3

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) hereby submits an amendment application for the Perry Nuclear Power Plant (PNPP). The proposed amendment would modify Technical Specification (TS) 3.6.2.2, "Suppression Pool Water Level," as well as TS surveillance requirements (SRs) 3.6.2.4.1 and 3.6.2.4.4 associated with TS 3.6.2.4, "Suppression Pool Makeup (SPMU) System" to allow installation of the reactor well to steam dryer storage pool gate in the UCP in MODEs 1, 2, and 3. The proposed amendment would also create a new Special Operations TS, TS 3.10.9, "Suppression Pool Makeup - MODE 3 Upper Containment Pool Drain-Down," to allow draining of the reactor well portion of the UCP in MODE 3. These proposed changes would permit the start of certain outage related activities in MODE 3.

An evaluation of the proposed amendment is enclosed. FENOC is requesting Nuclear Regulatory Commission (NRC) staff approval by February 24, 2017, and an implementation period of 60 days following issuance of the amendment. PNPP intends to expedite implementation of the required activities so that the amendment would be available for use in the PNPP's sixteenth refueling outage, beginning March 6, 2017.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 15, 2016.

Sincerely,

David B. Hamilton

Enclosure: Evaluation of Proposed License Amendment

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

Enclosure
L-16-083

Evaluation of Proposed License Amendment
(88 pages, excluding this page)

Evaluation of Proposed License Amendment

Subject: Proposed Revision to Modify Technical Specifications (TS) 3.6.2.2, 3.6.2.4, and Incorporate TS 3.10.9 to Allow Installation of the Reactor Well to Steam Dryer Storage Pool Gate in MODES 1, 2, and 3 and to Permit Upper Containment Pool Drain-Down in MODE 3.

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License NPF-58 for the Perry Nuclear Power Plant (PNPP).

The proposed amendment would allow installation of the reactor well to steam dryer storage pool gate in the upper containment pool (UCP) in MODES 1, 2, and 3, by revising the low water level limit in Technical Specification (TS) 3.6.2.2, "Suppression Pool Water Level," and revising two surveillance requirements in TS 3.6.2.4, "Suppression Pool Makeup (SPMU) System," to allow the gate installation and modify UCP and suppression pool level requirements.

The proposed amendment also creates Special Operations TS, TS 3.10.9, "Suppression Pool Makeup - MODE 3 Upper Containment Pool Drain-Down." This proposed TS allows the draining of the reactor well in MODE 3 as long as certain requirements are met. The acceptable range of combined UCP and suppression pool water levels are identified on a graph that would be provided as a figure in TS 3.10.9.

Presently, the installation of the reactor well to steam dryer storage pool gate and draining of the UCP are not allowed until MODE 4.

2.0 DETAILED DESCRIPTION

The proposed TS changes are as follows:

1. <u>TS 3.6.2.2:</u> A new suppression pool water level range ≥ 18 feet 3.2 inches and ≤ 18 feet 6 inches is established for conditions when the reactor well to steam dryer storage pool gate is installed in MODES 1, 2, and 3.

2. <u>TS SR 3.6.2.4.1:</u> Item c) to the upper containment pool water level surveillance requirement is added to maintain upper pool water level ≥ 23 feet 0 inches above the reactor pressure vessel (RPV) flange and the suppression pool water level ≥ 18 feet 3.2 inches when the reactor well to steam dryer storage pool gate is installed.

3. <u>TS SR 3.6.2.4.4:</u> A note to the surveillance requirement is added that eliminates the requirement to verify the upper containment pool gates are in the stored position or that they are otherwise removed from the upper containment pool if the upper containment pool water levels are maintained per TS SR 3.6.2.4.1.c, no work is being performed that has the potential to drain the upper fuel transfer pool, IFTS carriage is located in the upper pool, and the IFTS transfer tube shutoff valve 1F42F002 is closed.

4. <u>TS 3.10.9</u>: TS 3.10.9, "Suppression Pool Makeup – MODE 3 Upper Containment Pool Drain-Down," is created to suspend TS 3.6.2.2, "Suppression Pool Water Level," and TS 3.6.2.4, "Suppression Pool Makeup (SPMU) System," and to allow draining of the reactor well portion of the UCP in Mode 3 under

certain conditions. This newly added TS contains the LCO, APPLICABILITY, ACTIONS, and SURVEILLANCE REQUIREMENTS.

The proposed changes to the Technical Specifications are reflected in the annotated TS pages provided in Attachment 1. Retyped TS pages are provided in Attachment 2 for information only. Associated changes to the TS Bases are indicated in Attachment 3. The proposed TS Bases changes are provided for information only and will be controlled by TS 5.5.11 "Technical Specifications (TS) Bases Control Program." The analyses supporting the proposed changes were performed using the GOTHIC (Generation of Thermal-Hydraulic Information for Containments) computer program. A description of the GOTHIC model and event analyses is discussed in Attachment 4.

Figure 1 shows the arrangement of the UCP and UCP gates. Figure 2 shows the arrangement of the suppression pool.

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 Description of Plant

PNPP is a boiling water reactor (BWR/6) with a Mark III containment. The containment vessel encloses the drywell. The drywell is a cylindrical, reinforced concrete structure with a removable head. The UCP is illustrated in Figure 1. The reactor well portion of the UCP is located above the drywell head. The drywell encloses the reactor vessel and the reactor coolant system. It is designed to withstand the pressure and temperature of the steam generated by a reactor coolant system pipe rupture, and channels the steam to the suppression pool via horizontal vents located in the drywell wall. The suppression pool, illustrated in Figure 2, contains a large volume of water, which is available to receive and rapidly condense the steam discharged from a postulated design basis accident (DBA) pipe break in the reactor coolant system. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve (S/RV) discharges or from a loss of coolant accident (LOCA). The suppression pool must also condense steam from the reactor core isolation cooling (RCIC) system turbine exhaust and provides the main emergency water supply source for the reactor vessel.

The UCP provides water to the suppression pool following a LOCA by means of the suppression pool makeup (SPMU) system. The SPMU system consists of two 100 percent capacity lines. Each one directs a portion of the UCP water to the suppression pool by gravity when two normally closed valves in each line are automatically opened in response to either a low-low suppression pool water level signal or a timer set for 30 minutes after the LOCA begins.

PNPP TS 3.6.2.2 specifies the limits on the minimum and maximum water levels in the suppression pool. The minimum suppression pool water level limit ensures adequate coverage of the horizontal drywell vents, which maintains the pressure suppression

function when steam is discharged from the S/RV quenchers, the horizontal drywell vents, and the RCIC turbine exhaust lines. The minimum level also ensures net positive suction head (NPSH) is available for emergency core cooling system (ECCS) pumps that take suction from the suppression pool. As the ECCS flow is injected into the reactor pressure vessel and containment (as spray), some of the inventory becomes entrapped in plant locations that prevent the return of the flow back to the suppression pool. The minimum level specified by TSs ensures, during draw down by the ECCS pumps, that the top drywell vents in the suppression pool remain submerged by 2 feet to maintain the design-basis functions of the system.

The suppression pool volume plus the makeup volume from the upper pool is adequate to supply all possible post-accident entrapment volumes for suppression pool water, and keep the suppression pool at an acceptable water level. In order to ensure the proper amount of water is transferred (that is, dumped) to the suppression pool, TS SR 3.6.2.4.4 requires that the reactor well to steam dryer storage pool gate must be removed (or placed in its stored position) when in MODES 1, 2, and 3. The water supplied by the SPMU system, together with the water inventory in the suppression pool, is sufficient to supply all post-accident entrapment volumes and keep the suppression pool at an acceptable level (2 feet above the top of the horizontal vents) to condense steam exiting the vents.

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

3.1.2 Description of Current Technical Specification Requirements

TS 3.6.2.2 requires the suppression pool water level to be maintained greater than or equal to 17 feet 9.5 inches and less than or equal to 18 feet 6 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 with a completion time of 2 hours. Otherwise, the plant is required to be in MODE 3 in 12 hours and MODE 4, "Cold Shutdown," in 36 hours. TS SR 3.6.2.2.1 requires verification every 24 hours that the suppression pool water level is within limits.

TS 3.6.2.4 requires two SPMU subsystems to be operable in MODES 1, 2, and 3. If the combined UCP and suppression pool water levels are not within limits, Required Action A.1 mandates that the level be restored to within limits in 4 hours. If the UCP water temperature is not within limits, Required Action B.1 mandates that the temperature be restored to within limits in 24 hours. If one SPMU subsystem is inoperable for reasons other than Conditions A and B, Required Action C.1 requires that the SPMU subsystem be restored to OPERABLE status within 7 days. If the required actions cannot be met, the plant is then required to be in MODE 3 in 12 hours and in MODE 4 in 36 hours.

The SRs for TS 3.6.2.4 confirm that the LCO is met as summarized below.

- SR 3.6.2.4.1 requires verification every 24 hours that the UCP water level is greater than or equal to 22 feet 9 inches, or greater than or equal to 22 feet 5 inches with the suppression pool water level greater than or equal to 17 feet 11.7 inches.

- SR 3.6.2.4.2 requires verification every 24 hours that UCP water temperature is less than or equal to 110 degrees Fahrenheit (°F).

- SR 3.6.2.4.3 requires verification every 31 days that each SPMU subsystem manual, power-operated, and automatic valve that is not locked, sealed, or otherwise secured in position is in the correction position.

- SR 3.6.2.4.4 requires verification every 31 days that all required UCP gates are in the stored position or are otherwise removed from the UCP.

- SR 3.6.2.4.5 requires verification every 24 months that each SPMU subsystem automatic valve actuates to the correct position on an actual or simulated automatic initiation signal.

3.1.3 Description of Current Technical Bases

The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from S/RV discharges or from a LOCA. The suppression pool must also condense steam from the RCIC system turbine exhaust and provides the main emergency water supply source for the reactor vessel.

The maximum suppression pool water level limit ensures that, following a LOCA, post-LOCA suppression pool swell loads, S/RV clearing loads, and other hydrodynamic loads are within design limits. In addition, maximum limits on the suppression pool water level will minimize the overflow into the drywell in the event of an inadvertent draining of the UCP into the suppression pool.

The minimum suppression pool water level limit ensures that a sufficient amount of water is available to adequately condense the steam by maintaining a 2 foot minimum post-LOCA horizontal vent coverage following the ECCS drawdown of water level in the suppression pool. The suction of the ECCS pumps is taken from the suppression pool. The drawdown of water level results from ECCS injection that becomes entrapped or held up in locations within the plant that do not drain back to the suppression pool. In addition, minimum limits on the suppression pool water level ensure adequate ECCS pump NPSH and sufficient heat sink capability for the decay and sensible heat released during a reactor blowdown from S/RV discharges or from a LOCA.

The function of the SPMU system is to transfer water from the UCP to the suppression pool after a LOCA to offset any ECCS injection flow that has become held up in the regions of the plant that do not drain back to the suppression pool. The water transfer from the SPMU system ensures a post-LOCA suppression pool coverage of greater

than or equal to 2 feet above the top of the horizontal vents, adequate suppression pool heat sink volume, and adequate NPSH for the ECCS pumps.

3.1.4 Description of Analysis

The proposed license amendment would permit PNPP to install the reactor well to steam dryer storage pool gate in MODES 1, 2, and 3, and drain the reactor well in MODE 3. Installation of the gate will isolate water in the steam dryer storage pool from the SPMU system. To compensate for the lost inventory, the in-place suppression pool volume must be increased to a level that is slightly below the existing high water level (HLW) limit. To compensate for the reduction in UCP volume due to draining of the reactor well in MODE 3, the in-place suppression pool volume must be increased above the current HWL limit. The plant response resulting from the draining of the UCP and the increase in the suppression pool water level has been re-analyzed at the defined MODE 3 reactor conditions.

The new analysis was 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 multi-compartment, 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 heating and ventilation analyses at other nuclear power plants. The GOTHIC code has not been previously used at PNPP to model containment response to a design basis accident (DBA). Further discussion of the GOTHIC code and the PNPP model is contained in Attachment 4.

The primary reason for the UCP and suppression pool water level requirements is to ensure that the required post-accident water inventory is available. Sufficient water inventory in the suppression pool is required to provide ample coverage of the drywell-to-containment horizontal vents to maintain the pressure suppression function of containment. The suppression pool inventory at the minimum drawdown also ensures NPSH for the 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 installations in MODES 1, 2, and 3, and draining of the UCP in MODE 3.

The proposed changes include:

1) A revised operating range for the UCP and suppression pool water levels that will be in effect during the time period that the reactor well to steam dryer storage pool gate is installed in MODES 1, 2, and 3.

2) A revised operating range for the suppression pool water level as a function of the UCP water level that will be in effect during the time period that the UCP is being drained in MODE 3.

During draining of the reactor well, the UCP level will be below the current TS limit and the suppression pool will be above the current TS HWL limit. The increase in the suppression pool level has the potential to impact the post-LOCA containment thermal-hydraulic response and hydrodynamic loads. Analyses have been performed to evaluate the SPMU licensing basis events under MODE 3 drain-down conditions. These analyses assess the impact of the containment response on the hydrodynamic loads, the long term heat sink capability of the plant, and the maximum permissible steam leakage from drywell directly to the containment air-space as a result of drywell bypass leakage.

The analyses were performed to support assumed plant conditions that include the reactor steam dome pressure ≤ 230 psig and the reactor subcritical for ≥ 2 hours. These and other plant conditions, as described in the following sections, need to be met before UCP drain-down can begin and are incorporated in the new TS 3.10.9 for UCP drain-down in MODE 3. This will ensure that the reactor conditions during drain-down are bounded by the supporting analyses. The proposed reactor steam dome pressure limit of 230 psig includes a 5 psi margin to the analyzed value of 235 psig.

3.1.5 Impact of Proposed Change on Suppression Pool Makeup Design Basis

PNPP USAR Section 6.2.7.1 provides the design bases for the SPMU system. Twelve criteria are identified as being used in the design of the SPMU system. The effect of the proposed amendment on each of the design basis criteria listed is identified as follows:

a. The [suppression pool makeup] system is redundant with two 100 percent capacity lines. The redundant lines are physically separated and all electrical power and control is separated into two divisions in accordance with IEEE [Institute of Electrical and Electronic Engineers] Standard 279.

   The physical design of the system will remain unaltered, as there will be no physical changes made to any plant components or structures. Therefore, this design basis criterion will remain unaffected and will not change.

b. The [suppression pool makeup] system is Safety Class 2, Seismic Category 1.

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

c. The minimum long term post-accident suppression pool water coverage over the top of the top drywell vent is 2'-0".

   This design basis criterion will not change. The proposed changes to the suppression pool and UCP water levels are based on maintaining the water coverage over the top row of drywell vents at greater than or equal to 2 feet.

d. The minimum normal operation low water level (LWL) suppression pool height above the top drywell vent centerline is 6'-9.5".

If the reactor well to steam dryer storage pool gate is installed in MODEs 1, 2, or 3, the suppression pool LWL will need to be increased, as discussed in section 3.2.1 of this evaluation. If MODE 3 drain-down of the reactor well is pursued, the suppression pool LWL will need to be increased in accordance with proposed TS Figure 3.10.9-1, as discussed in sections 3.2.2 and 3.2.3 of this evaluation.

e. The maximum normal operation high water level (HWL) suppression pool height above the top drywell vent centerline is 7'-6".

   This design basis criterion remains unaffected if the reactor well to steam dryer storage pool gate is installed in MODEs 1, 2, or 3. However, this design basis criterion will change if MODE 3 drain-down of the reactor well is pursued. In this case, the suppression pool HWL will need to be increased in accordance with proposed TS Figure 3.10.9-1, as discussed in sections 3.2.2 and 3.2.3 of this evaluation.

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

   For MODE 3 drain-down of the reactor well, TS Figure 3.10.9-1 is proposed, which changes the required amount of water in the suppression pool as a function of the water level that exists in the UCP. Having adequate suppression pool water inventory is addressed throughout the technical evaluation section of this LAR (section 3.0). Although suppression pool water level has been evaluated near the top of the drywell weir wall, an upper (HWL) limit on the drain-down operating band is identified in proposed TS Figure 3.10.9-1 to limit the amount of water added to the suppression pool during the drain-down evolution. This will help to minimize the effects of the condition where added water dumped to the suppression pool from the UCP pool might over flow the drywell weir wall. This potential occurrence is already recognized in USAR sections 6.2.7.1, 6.2.7.2, and 6.2.7.3.3. The effects of potential drywell flooding with inadvertent UCP dump is discussed in section 3.8.1 of this evaluation. This design basis criterion remains satisfied by the proposed amendment.

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

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

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

3. Volume in the steam lines out to the first MSIV for three lines and out to the second MSIV on one line.

4. An allowance for containment spray holdup on equipment and structural surfaces.

   Adjustments to the existing entrapment volumes apply to installation of the UCP gate in MODES 1, 2, and 3 and to the MODE 3 drain-down evolution. Additional entrapment volumes in the UCP during the MODE 3 drain-down evolution are also considered. These alterations represent changes to the existing entrapment volumes identified in this USAR design basis criterion. Each of the individual entrapment volume alterations and additions is discussed in sections 3.2.1 and 3.2.2 of this evaluation.

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

   This design basis criterion is maintained and will not change.

i. Piping components which would be wetted in the event of a drywell flooding transient (from inadvertent dump of the upper pool (SPMU) with the suppression pool at maximum operating level and under negative drywell pressure conditions) have been analyzed and would not result in adverse consequences to these components.

   The amount of freeboard distance available from the existing suppression pool HWL to the top of the drywell weir wall will be reduced, as water level after the drain-down evolution will be near the top of the weir wall. To minimize the potential impact of drywell flooding during/after an upper pool dump, an administrative requirement will be imposed during drain-down to maintain a zero to positive drywell to containment differential pressure, as discussed in section 3.8.1 of this evaluation. This will help to ensure that this design basis criterion is maintained.

j. The minimum normal operation suppression pool volume at LWL is adequate to act as a short term energy sink without credit for upper pool dump.

   This design basis criterion has not changed.

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

   This design basis criterion is maintained and will not change.

l. The system gravity dump time through one of the two redundant lines is less than or equal to the minimum pump time; pump time is determined by dividing pumping

volume (upper pool makeup volume plus volume in the suppression pool stored between LLWL [low low water level] and minimum top vent coverage) by the total maximum runout flow rate from all five ECCS pumps.

This design basis criterion has not changed and has been re-evaluated for the new suppression pool and UCP water levels. As discussed in section 3.8.2 of this evaluation, this criterion is met for both UCP gate installation in MODES 1, 2, and 3 and for reactor well drain-down in MODE 3.

In summary, all of the above listed design basis criteria for the SPMU system are maintained, except items d, e and g. The suppression pool LWL will be increased above its current limit when the UCP gate is installed and when the UCP drain-down evolution is performed (item d). The suppression pool HWL will be increased above its current limit when the UCP drain-down evolution is performed (item e). Adjustments to the existing DBA entrapment volumes are needed for gate installation and UCP drain-down (item g), as discussed in sections 3.2.1 and 3.2.2 of this evaluation.

3.2 Post-Accident Vent Coverage

The initial minimum suppression pool water level ensures adequate vent coverage during the first minutes following a LOCA. As the ECCS systems drawdown the level in the suppression pool and inject water into the RPV, the spillage through the line break collects in the bottom of the drywell. 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 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 (SPMU volume) accounts for the minimum suppression pool level (after filling the required entrapment volumes) to assure that the suppression pool water level will provide at least 2 feet of coverage above the top of the top row vents.

The current design-basis of the required makeup volume considers four primary entrapped volumes.

a) The drywell free volume inside and below the top of the drywell weir wall [44,673 cubic feet (ft^3)];

b) The added water volume needed to fill the vessel from a condition of normal power operation to a post-accident complete fill of the vessel including top dome (10,546 ft^3);

c) Volume in the steam lines out to the first main steam isolation valve (MSIV) for three lines and out to the second MSIV for one line (963 ft^3);

d) An allowance for containment spray holdup on equipment and structural surfaces (1,500 ft^3).

These four volumes are considered in the design-basis makeup requirements, where the total entrapment volume is estimated to be 57,682 ft^3. This total entrapment volume contains margins when considering an accident during the proposed operating conditions.

3.2.1 Reactor Well Gate Installation (MODES 1, 2, and 3)

Adjustments to the existing entrapment volumes are considered for installation of the reactor well to steam dryer storage pool gate during MODES 1, 2, and 3. The basis of the required makeup volumes considers the following primary entrapped volumes.

a) The free volume inside and below the top of the drywell weir wall (referred to hereafter as the drywell pool) is a fixed volume and must be accounted for during a design-basis large break of a main steam line or a recirculation suction line in MODES 1, 2, and 3. This entrapment volume is maintained for gate installation in MODES 1, 2, and 3.

b) The reactor pressure vessel (RPV) refill design-basis entrapment volume includes an allowance 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. This entrapment consideration is maintained for gate installation in MODES 1, 2, and 3.

   The RPV design-basis entrapment volume also accounts for post-accident ECCS flooding of the RPV from normal water level to the top of the steam dome. For gate installation in MODES 1, 2, and 3, the RPV entrapment volume is modified to consider flooding from the normal vessel water level to the centerline elevation of the main steam line (MSL) nozzles. For large main steam line breaks (MSLBs) and recirculation suction line breaks (RSLBs), the RPV is not expected to fill above the MSLs. ECCS flow will pour out the break and prevent the vessel from filling to no higher than the elevation of the MSLs. The entire volume of the MSLs is still considered an entrapment volume out to the inboard MSIVs in three lines, and out to the outboard MSIV in a single line. Any break flow that leaves the RPV will be trapped within the drywell pool until it floods over the weir wall, which is also still considered an entrapment volume for all modes of operation. Therefore, post-accident entrapment to the steam dome is replaced with post-accident entrapment to the centerline elevation of the MSL nozzles in consideration of large line breaks.

   For a small line break especially high in the RPV, the RPV may fill to the vessel dome if the ECCS flow is unrestrained. During a post-accident break, the current emergency operating procedures (EOPs) direct operators to maintain reactor vessel water level below Level 8 (that is, high reactor vessel water level), which is well below the elevation of the MSLs. The break flow associated with a small line break is not expected to challenge the operator's ability to maintain water level below Level 8, thereby preventing ECCS flow from filling the vessel above

the steam lines to the steam dome. In the event operators declare the Level 8 instrument reading unavailable due to suspected reference leg boiling, EOPs instruct operators to open the main steam S/RVs (dump to suppression pool) and increase the ECCS injection to the RPV. Under these conditions, the ECCS injection will flow out the main steam S/RVs, thereby limiting the RPV flooding to the elevation of the MSL nozzles.

Consequently, a RPV entrapment volume that considers flooding from normal water level to the centerline elevation of the MSL nozzles is sufficient to address the expected volume applicable during large and small line ruptures. This is a change from the current licensing basis entrapment volume.

c) The MSL volume may flood or partially fill during a MSLB event. The flooding of the MSLs is also considered an entrapment volume for reactor well gate installation during MODES 1, 2, and 3.

d) The holdup allowance for containment spray on equipment and structural surfaces is maintained for reactor well gate installation during MODES 1, 2, and 3. Containment spray entrapment volume in the UCPs is not considered since the pool level will remain at the UCP normal water level (that is, 3 inches higher than the UCP minimum level) until drain-down in MODE 3 at reduced reactor pressure. The UCP normal water level is above the reactor well to steam dryer storage pool partition wall, thereby eliminating the steam dryer storage pool and the fuel transfer pool as additional entrapment volume areas.

The design-basis total entrapment volume (57,682 ft^3) is reduced by 4,112 ft^3 in the analysis that evaluates gate installation in MODES 1, 2, and 3. The reduced total entrapment volume applicable when the UCP gate is installed is 53,570 ft^3. This is a result of a reduction in the RPV entrapment volume when replacing the conservatively assumed flooding of the RPV to the steam dome with a more realistic flooding of the RPV to the centerline elevation of the MSL nozzles.

The design-basis SPMU inventory available for transfer to the suppression pool (32,573 ft^3) is reduced when the reactor well to steam dryer storage pool gate is installed. Gate installation prevents the water inventory in the dryer pool from entering the reactor well and then flowing over the steam separator storage pool weir wall to where the SPMU system dump lines are located. The SPMU inventory that is isolated by the gate corresponds to a liquid volume of 7,472 ft^3, which represents the volume in the steam dryer storage pool at the current UCP TS minimum pool water level above the elevation of the steam separator storage pool weir wall. For conservatism, an additional 105 ft^3 is added to the SPMU lost inventory (with the gate installed) to ensure consistency with the current design-basis SPMU dump inventory. The total SPMU lost inventory (reduction) is 7,577 ft^3 (7,472 ft^3 + 105 ft^3).

This SPMU inventory reduction (7,577 ft^3) with the gate installed in MODES 1, 2, and 3 is not fully offset by the reduction in the RPV entrapment volume (4,112 ft^3).

Consequently, additional pool inventory is required prior to installing the gate in MODES 1, 2, and 3. To offset a portion of the SPMU inventory reduction caused by gate installation, the proposed TS change maintains the upper containment pool at the normal upper pool operating level of 688 feet 7 inches (23 feet above the RPV flange) until the start of the reactor well drain-down process. This represents an UCP level that is 3 inches above the current UCP TS minimum level of 688 feet 4 inches. The additional inventory available at this level decreases the SPMU inventory reduction from 7,577 ft^3 to 6,748 ft^3.

The remaining inventory needed to ensure the SPMU design basis criterion of 2-foot post-accident vent submergence is defined by the difference between the SPMU system inventory reduction (6,748 ft^3) and the change in the RPV entrapment volume (4,112 ft^3). This reduces the shortage to 2,636 ft^3 of necessary volume. To offset the remaining SPMU shortage due to gate installation, the proposed TS change increases the suppression pool water level above the current TS LWL (17 feet 9.5 inches) prior to gate installation. The required increase in the suppression pool water level to offset the needed volume of 2,636 ft^3 is 5.4 inches above the TS LWL. This corresponds to a new analytical lower limit of 18 feet 2.9 inches (17 feet 9.5 inches + 5.4 inches) when the reactor well to steam dryer storage pool gate is installed in MODES 1, 2, and 3.

The analyses for UCP gate installation utilized a specific uncertainty value for the suppression pool level instrumentation. The uncertainty value was accurately determined, so as to avoid overly conservative assumptions that would reduce suppression pool level operating margin. Accounting for a level measurement uncertainty of ± 0.3 inches (applicable to the existing level instrumentation) results in a new minimum water level limit of 18 feet 3.2 inches (18 feet 2.9 inches + 0.3 inches) with the gate installed in MODES 1, 2, and 3. The current TS HWL is 18 feet 6 inches, which allows 2.8 inches of operating band (18 feet 6 inches - 18 feet 3.2 inches). Even though the operating band is reduced during operations in MODES 1, 2, and 3 with the gate installed, the suppression pool water level can be managed by plant operations during the short duration that this condition will exist.

Figure 2 shows the current and proposed suppression pool water level limits as well as other important suppression pool elevations for gate installation in MODES 1, 2, and 3. Figure 1 shows the arrangement of the UCP and the proposed UCP water level for gate installation in MODES 1, 2, and 3. Maintaining the level of the suppression pool and the UCP, as specified for gate installation, will ensure that the design basis 2-foot post-accident vent submergence criterion is met.

3.2.2 Reactor Well Drained in MODE 3

Adjustments to the existing entrapment volumes and additional entrapment volumes in the UCP are considered for the MODE 3 drain-down of the reactor well. The basis of the required makeup volumes considers the following primary entrapped volumes.

a) The drywell pool is a fixed volume and must be accounted for during a design-basis large break of a main steam line or a recirculation suction line in MODE 3.

This entrapment volume is maintained for MODE 3 drain-down of the reactor well.

b) The RPV refill design-basis entrapment volume includes an allowance 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 condition (≤ 235 psig pressure), there is essentially no voiding below the water level and level shrink is substantially reduced. Therefore, compensating the level in the vessel for steam void and level shrink from MODE 1 conditions does not need to be considered.

The RPV design-basis (MODE 1) entrapment volume accounts for post-accident ECCS flooding from normal water level to the top of the steam dome. As discussed for the reactor well gate installation in MODES 1, 2, and 3, the RPV entrapment is modified to consider flooding from the normal vessel water level to the centerline elevation of the MSL nozzles. For large vessel line breaks (MSLB, RSLB), the RPV is not expected to fill above the MSLs. ECCS flow will pour out the break and prevent the vessel from filling to no higher than the elevation of the MSLs. The entire volume of the MSLs is still considered an entrapment volume out to the inboard MSIVs in three lines, and out to the outboard MSIV in a single line. Any break flow that leaves the RPV will be trapped within the drywell pool until it floods over the weir wall, which is considered an entrapment volume for all modes of operation. Therefore, post-accident entrapment to the steam dome is replaced with post-accident entrapment to the centerline elevation of the MSL nozzles for large line breaks.

For a small line break especially high in the RPV, the RPV may fill to the vessel dome if the ECCS flow is unrestrained. The EOPs direct operators to maintain reactor vessel water level below Level 8 (that is, high reactor vessel water level), which is well below the elevation of the MSLs. Under MODE 3 conditions with the plant already shutdown, the operators are largely focused on initiating and maintaining ECCS injection and RPV level control. Break flow associated with a small line break is not expected to challenge the operator's ability to maintain water level below Level 8, thereby preventing ECCS flow from filling the vessel above the steam lines to the steam dome. In the event operators declare the Level 8 instrument reading unavailable due to suspected reference leg boiling, EOPs instruct operators to open the main steam safety relief valves (dump to suppression pool) and increase the ECCS injection to the RPV. Under these conditions, the ECCS injection will flow out the main steam S/RVs, thereby limiting the RPV flooding to the elevation of the MSL nozzles.

Consequently, a RPV entrapment volume that considers flooding to the centerline elevation of the MSL nozzles is sufficient to address the expected entrapment volume applicable during large and small line ruptures while in MODE 3 hot shutdown conditions. This is a change from the current licensing basis entrapment volume.

c) The MSL volume may flood or partially fill during a MSLB event. The flooding of the MSLs is also considered an entrapment volume under MODE 3 drain-down conditions.

d) The holdup allowance for containment spray on equipment and structural surfaces is maintained for MODE 3 conditions. In addition, with the reactor well in the UCP drained as proposed in MODE 3, the reactor well becomes an entrapment volume under containment spray (CS) operations. The reactor well entrapment volume includes the water volume from the well floor to top of the steam separator storage pool weir wall (Figure 1). If level in the reactor well floods over the weir wall, this inventory collects in the steam separator storage pool and is made available to the SPMU dump system. The reactor well entrapment volume considers the displaced volume of the drywell head. Containment spray may also collect in the steam dryer storage and fuel transfer pools after drain-down. Therefore, an allowance for steam dryer/fuel transfer pool depth of 2 inches below the reactor well partition wall is considered.

   The CS system is not required and will not automatically start for a large line break LOCA at the defined MODE 3 conditions to limit containment pressure. However, operators are not prevented from manually actuating sprays.

e) If the gravity drain lines from the UCP are used to transfer water directly from the reactor well to the suppression pool, then the drain lines represent an entrapment volume if the lines become isolated during a LOCA event.

The design-basis total entrapment volume (57,682 ft^3) is increased by approximately 2,568 ft^3 in the analysis that evaluates MODE 3 drain-down conditions. The increased total entrapment volume applicable during MODE 3 drain-down conditions is 60,250 ft^3. The additional CS trapped in the UCP is partly offset by reducing the RPV entrapment from a scenario that fills to the vessel steam dome to a scenario that fills the vessel to the MSLs (with no void collapse).

To offset the total entrapment volume of 60,250 ft^3, a makeup volume of 6,412 ft^3 from the UCP (steam separator storage pool) is available. In addition, operating at the current suppression pool LWL TS limit provides 22,701 ft^3 of drawdown volume above the 2-foot vent submergence post-accident requirement. Therefore, the total inventory required above the LWL in MODE 3 to provide 2 feet of vent coverage is 31,137 ft^3. This represents the total entrapment volume reduced by the makeup volume and inventory available in the suppression pool while operating at the TS LWL.

The total inventory that needs to be added to the suppression pool under MODE 3 conditions corresponds to a lower bound analytical limit of 23 feet 1 inches above the suppression pool floor. The analyses for UCP drain-down conservatively assumed a 1-inch level instrument uncertainty for the suppression pool. This 1-inch whole number uncertainty value also enables the specified operating range to be identified in even

1-inch measurement values. By providing 1 inch of level measurement uncertainty and an operating range of 5 inches, the new suppression pool level range is ≥ 23 feet 2 inch and ≤ 23 feet 7 inches for operations in MODE 3 with the reactor well drained, the reactor pressure ≤ 235 psig (analytical value), and the reactor core subcritical for ≥ 2 hours. Containment loads were evaluated at a maximum suppression pool level of at least 23 feet 8 inches, which includes a measurement uncertainty of 1.0 inch above the operating band. The containment loads were determined to be acceptable for suppression pool level which is at least 5 feet 2 inches higher than the current suppression pool HWL limit (18 feet 6 inches) in MODE 3, when pressure is less than or equal to 235 psig (analytical value).

Figure 2 shows the current and proposed suppression pool water level limits as well as other important suppression pool elevations for gate installation in MODE 3 drain-down of the reactor well. Figure 1 shows the arrangement of the UCP and the proposed UCP water level for MODE 3 drain-down of the reactor well. Maintaining the level of the suppression pool, as specified for MODE 3 drain-down of the reactor well, will ensure that the design basis 2-foot post-accident vent submergence criterion is met.

3.2.3 Reactor Well Drain Evolution (MODE 3)

The reactor well drain evolution in MODE 3 can begin when the proposed TS 3.10.9 LCO requirements are met. These requirements include:

- Suppression pool average temperature ≤ 110 °F,
- UCP and suppression pool levels in compliance with the drain-down curve illustrated in Figure 3.10.9-1,
- Steam dryer storage pool and fuel transfer pool water levels are ≥ 22 feet 8 inches above the RPV flange,
- Reactor dome pressure ≤ 230 psig,
- Reactor has been subcritical for ≥ 2 hours,
- SPMU subsystem valves are operable based on current TS surveillance requirements,
- UCP temperature is ≤ 110 °F,
- No work is being performed that has the potential to drain the fuel transfer pool,
- IFTS carriage is located in the upper pool,
- IFTS transfer tube shutoff valve 1F42F002 is closed, and
- Reactor well to steam dryer storage pool gate is installed.

The reactor well drain-down evolution has been developed to ensure the SPMU design-basis functions are satisfied for post-accident entrapment volumes during MODE 3 drain-down of the reactor well (as described in Section 3.2.2). The drain-down process shall be controlled within the acceptable operating region illustrated in proposed TS Figure 3.10.9-1 The acceptable operating region is defined by the necessary upper containment pool and suppression pool water levels (in combination) that ensure compliance with the SPMU system design-basis criterion of 2 feet of drywell vent submergence following a postulated LOCA. The reactor well draining process may proceed by any means as long as both pool levels are maintained in accordance with

TS Figure 3.10.9-1. Specifically, the upper pool may be drained directly to the suppression pool, thereby preserving the necessary combined water inventory between the two pools for post-accident vent coverage. Alternatively, external sources of water may be used to maintain the suppression pool within the acceptable band while the upper pool is drained elsewhere.

The upper containment pool level shown in TS Figure 3.10.9-1 is defined for a range that spans from elevation 664'-7" at the reactor well floor (empty pool) to the elevation of the refueling deck at elevation 689'-6" (24 feet 11 inches above the reactor well floor) as shown in Figure 1. As the UCP level is drained below the top of the steam separator storage weir wall (680 feet 0 inches), the only available SPMU inventory is the inventory in the steam separator storage pool. Below the weir wall, the upper pool water level in TS Figure 3.10.9-1 corresponds to the water level in the reactor well. The post-accident entrapment volumes, as developed for MODE 3 drain-down of the reactor well, assumed that the reactor well becomes an entrapment volume during containment spray operations. However, at the start of the drain-down sequence, the reactor well remains full. Consequently, the initial suppression pool level to be maintained at the start of the drain-down sequence need not consider the reactor well as an entrapment in order to ensure post-accident vent submergence. The lower bound drain curve in TS Figure 3.10.9-1 was developed assuming the inventory within the reactor well is transferred directly to the suppression pool (or alternatively managed by external makeup sources). In this way, the reactor well entrapment volume is offset by the volume of the well added directly to the suppression pool. When the reactor well is fully drained, the required suppression pool minimum water level from the drain-down curve matches the necessary inventory determined by the entrapment calculations discussed in Section 3.2.2.

If during the drain-down evolution of the reactor well, operators decide to drain a portion of the reactor well elsewhere, staying within the acceptable operating region for pool levels as identified in TS Figure 3.10.9-1 remains imposed until MODE 4. Alternate external sources for makeup to the suppression pool, such as the condensate storage tank, are available to provide the necessary water inventory to the suppression pool to maintain the minimum required inventory.

The upper bound drain-down curve (TS Figure 3.10.9-1) provides a 5-inch band for operators to perform the drain-down evolution. The upper band minimizes the potential for flooding over the drywell weir wall if the UCP were to inadvertently dump to the suppression pool (discussed in Section 3.8.1).

Maintaining pool levels within the lower and upper bound limits defined by TS Figure 3.10.9-1 will ensure that the suppression pool water level will provide at least 2 feet of coverage above the top row of vents during draining and after the reactor well has been drained in MODE 3.

3.3 Hydrodynamic Loads

The proposed MODE 3 drain-down operation (TS 3.10.9) requires raising the water level in the suppression pool above the current TS HWL 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 in excess of 5 feet 2 inches above the current TS high water level limit of 18 feet 6 inches. The evaluation shows that the hydrodynamic loads imparted with the revised water level and reactor pressure less than or equal to 235 psig will be bounded by those from a DBA (at full power operating pressure) with the suppression pool filled to the current high water level limit. For this evaluation, the term DBA is defined as either the large break RSLB or 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 Reference 1 was considered.

The containment loads generated during the first part of a LOCA are primarily a function of the drywell pressure rise and secondarily by a function of the suppression pool water level, temperature, and other parameters. A GOTHIC analysis of the limiting LOCA in MODE 3 (MSLB), with the primary system at 235 psig following two hours of post-shutdown decay and with the suppression pool water level near the top of the drywell weir wall (that is, in excess of 5 feet 2 inches above the current HWL limit), calculated a peak drywell pressure of 13.8 psig (28.53 psia) (as indicated in Figure 9 in Attachment 4). This compares with a peak drywell pressure of 23.7 psig (38.38 psia) for a GOTHIC analysis of the design-basis MODE 1 MSLB accident with the suppression pool level at the current TS HWL limit of 18 feet 6 inches. Consequently, the containment loads at the MODE 3 drain-down conditions are bounded by the licensing DBAs evaluated at full power operating pressure. The difference in drywell pressure and its effects have been evaluated against the effects of increased suppression pool water level at the MODE 3 low pressure conditions for each of the hydrodynamic loads, as follows.

3.3.1 Water Jet Loads

During the vent clearing transient, the weir wall, the weir side of the drywell wall and the LOCA vents will experience drag loads due to the suppression pool water being forced through the weir annulus and the LOCA vents by the rising drywell pressure. 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 act later in the transient (Reference 1).

All of these loads are primarily a function of the drywell pressure. For the low pressure LOCA in MODE 3, the loads will be smaller than the DBA loads due to the reduced drywell pressure. The increased liquid level in the suppression pool is fully accounted for in the drywell pressure and will have no additional impact on these loads.

3.3.2 LOCA Air Bubble Loads

As the air from the drywell is forced through the vents to the suppression pool and air rises through the pool, differential pressures are imposed on the weir wall, drywell wall, containment wall, and base mat. 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 in MODE 3 than for the DBA and the associated loads will be smaller for the low pressure events. As discussed below, pool bubble dynamics in the suppression pool are expected to be less severe for the lower pressure events and therefore produce lower pressure differentials.

3.3.3 Pool Swell Drag and Impact Loads

During a LOCA, the air in the drywell is forced into the suppression pool through the vents. The air forms a bubble in the pool, lifting 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 in the rising pool.

The impact loads are a function of the pool surface velocity. The pool swell rate 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. For the low pressure LOCA, the drywell pressure feeding the bubble is smaller than the DBA drywell pressure and the resultant vent flow rates are smaller. The ultimate bubble size in a low pressure LOCA will be smaller than that experienced during a DBA and the rate of bubble growth will be smaller, resulting in smaller swell velocity and lower impact and drag loads. Further, as indicated in Reference 4 (page 3-3), pool swell velocity is substantially reduced (factor of 2) if there is venting through only one row of vents as opposed to two rows of vents. For the low pressure event in MODE 3, the GOTHIC analysis predicts that most of the venting will occur through the top row of vents with only a brief period of flow passing through the second row of vents and none through the bottom row of vents.

Therefore, the pool swell and impact loads associated with a LOCA at reduced pressure during MODE 3 with an increased suppression pool water level will be less than that experienced during the full-power MODE 1 DBA.

3.3.4 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. During the low pressure LOCA in MODE 3 the pool swell is smaller than that experienced for the DBA. Therefore, the maximum velocity of the falling water, impact loads, and drag loads will be smaller in the low pressure events.

3.3.5 Froth Impingement and Drag Loads

When the bubble breaks though 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 events as compared to the DBA, there will be less froth and the maximum froth level will be lower during the low pressure events. Therefore, these loads are all bounded by the DBA.

3.3.6 Condensation Oscillation Loads

Once the vents have cleared and the pool swell transient has passed, there may be a period where the surface of the steam bubble that forms in the pool just beyond the top vent oscillates at low frequencies (3-20 Hz) causing cyclic loading on submerged pool structures and boundaries. This condensation induced oscillation mode (CO) can occur at steam flow rates between 10 and 25 pounds mass per second per square foot (lbm/ft^2-s) (Reference 4, page 3-28).

CO can occur in the low pressure LOCA as well as the DBA and intermediate and small break accidents when the vent flows are in the critical range. The magnitude and frequency of the CO loads are functions of the pool temperature, vent steam/air flows, and air mass fraction flowing through the vent. General Electric (GE) developed load and frequency functions that consider these effects and are bounding for all break events. The pool temperatures, vent flows, and air mass fractions for the low pressure LOCA are within the range of these parameters for the considered breaks. Further, the strength of the pressure pulses at the upper vent is independent of the vent submergence (Reference 1) and vent submergence is not a parameter in the GE methodology accepted by the NRC for calculating CO loads on equipment. Therefore, the CO load function will be bounding for the low pressure LOCA. From the GOTHIC analysis of the low pressure MSLB event, the vent steam/air flow falls below 10 lbm/ft^2-s at approximately 55.8 seconds after the break initiation and remains below that value through the remainder of the transient. In view of the short duration for CO potential during the low pressure LOCA and the conservatism built into the GE load and frequency functions, this increase in load is not considered to be significant.

3.3.7 Chugging Loads

When the steam flow through the top vents falls below 10 lbm/ft^2-s, the oscillatory condensation turns to an erratic chugging mode with a pressure pulse generated. The Nuclear Regulatory Commission (NRC) has determined that loads analysis for chugging for the DBA cover all break sizes (Reference 4). Chugging in the low pressure LOCA is not expected to be significantly different from other cases previously considered. The vent submergence was not identified as a significant parameter affecting chugging loads and is not used in the method recommended by GE for calculating chugging loads (Reference 1).

Humphrey Concern 19.1 identified that increased vent submergence could cause an increase in chugging loads. A technical evaluation was performed, as follows.

The evaluation considered a vent submergence up to 12 feet and determined that, in general, the GE load definition enveloped the increased chugging loads. Localized loads in the frequency range between 15 and 32 Hz may exceed the load definition. The exceedances were deemed acceptable since they represented either a small percentage overload or were applied to the basemat. The exceedance on the basemat is not of any consequence since the hydrostatic head ensures that a negative pressure will not be imposed on the liner and there are no natural modes of vibration that are excitable. The NRC staff accepted this response as documented in the Perry Safety Evaluation Report (SER), Reference 5. The NRC SER disposition for Humphrey Concern 19.1 (12 foot submergence due to upper pool dump) envelopes the chugging loads for the proposed increase in the suppression pool level of 5 feet 2 inches above the HWL (due to MODE 3 drain-down) and the potential level at the top of the weir wall (due to upper pool dump).

3.3.8 Drywell Depressurization Loads

When ECCS water is eventually injected into the vessel, it may spill into the drywell and condense the steam, depressurizing the drywell. This results in inward loads on the drywell wall. The low pressure in the drywell can cause the flow through the vents to the suppression pool to come back into the weir annulus and up through the annulus into the drywell introducing potential jet impingement, impact, and drag loads.

All of these loads are primarily a function of the 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 is at the suppression pool temperature to maximize the containment pressure. For the low pressure (drain-down) events in MODE 3, the energy deposited to the suppression pool will be less than in the DBA and the suppression pool temperature will be lower. This will result in a lower containment pressure and therefore lower depressurization loads.

3.3.9 Safety/Relief Valve Actuation Loads

Hydrodynamic loads from the S/RV actuation are partially dependent on discharge leg submergence. The loads, however, are far more dependent on reactor 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 with the resolution of Brookhaven National Laboratory concerns BNL-2 and BNL-3 as documented in the Perry Safety Evaluation Report (SER), Reference 5. The SER indicated that the S/RV discharge line thrust loads, which would result from S/RV actuation at elevated suppression pool levels, are within the capability of the existing S/RV discharge line configurations. Therefore, with the vessel pressure less than or equal to 230 psig during MODE 3 drain-down conditions, the loads from an S/RV lift will be significantly less than the design values.

3.4 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 net positive suction head (NPSH) requirements in the Perry USAR (References 2 and 3). The analyses are performed assuming 212 °F suppression pool temperature (clean strainer) and 185 °F (full loaded strainer), design pump runout flows, and atmospheric conditions. The USAR analyses show that adequate NPSH is available with the suppression pool level at the minimum drawdown elevation of 14 feet 2 inches above of 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 feet 2 inches) is protected. Further, the long-term GOTHIC simulations demonstrate that suppression pool temperatures under MODE 3 post-accident conditions remain below the design basis limits. Therefore, there are no concerns regarding ECCS pump NPSH requirements as a result of these changes.

3.5 Long-Term Heat Sink

The suppression pool volume provides a long-term heat sink for the decay and sensible heat released during a LOCA. The suppression pool volume required for MODE 3 drain-down conditions will be increased to offset the reduction in SPMU system dump volume when the reactor well is drained. The combined water inventory between the UCP and the suppression pool will therefore be maintained during the drain-down of the reactor well in MODE 3 to ensure that a minimum heat sink inventory in the suppression pool during a design-basis event is sufficient to provide the necessary long-term heat removal. Thus, the heat sink volume available under the proposed MODE 3 drain-down conditions provides the necessary post-accident heat sink to ensure the long-term suppression pool temperatures remain within limits (that is, 185 °F), and consequentially, the containment air pressure and temperature remain within limits.

GOTHIC simulations were performed considering the MSLB and RSLB design-basis accidents with the reactor well drained, reactor pressure equal to 235 psig, initial suppression pool water levels at the current TS LWL (17 feet 9.5 inches), initial suppression pool temperatures ≤ 110 °F, and minimum ECCS operations. The initial suppression pool water level in the GOTHIC simulations (for the long-term events) was conservatively evaluated at the current TS LWL (including a conservative UCP dump volume), although initial suppression pool levels for drain-down in MODE 3 will be higher as defined by the pool level requirements illustrated in proposed TS Figure 3.10.9-1. The analyses show a peak long-term suppression pool temperature of approximately 173 °F (RSLB event), which maintains a 12 °F margin between the calculated pool temperature and the design temperature limit of 185 °F for the suppression pool (Attachment 4, Figure 12).

3.6 Personnel and LOCA Dose Analysis

3.6.1 Personnel Dose

Based on the survey data from the refuel floor in MODE 4 and expected conditions in MODE 3, the resultant dose rate from the reactor in MODES 3 and 4 are the same. The only physical difference between MODE 3 and MODE 4 is the reactor coolant temperature. Consequently, the drain-down of the upper pool reactor well in either MODE 3 or in MODE 4 will produce the same dose results.

3.6.2 LOCA Dose

The PNPP DBA LOCA dose calculation assumes that the event is initiated from hot full power, MODE 1 conditions. While the containment pressure does not reach the automatic containment spray initiation setpoint, this analysis credits manual operation of the containment spray (CS) system for fission product scrubbing in containment. The GOTHIC analysis of DBA event conditions occurring during the proposed drain-down mode of operation in MODE 3 will result in containment pressure that remains below the 8.0 psig containment pressure setpoint (9.0 psig analytical value) for automatic spray initiation. The PNPP EOP requires that operators initiate containment spray as one of the first steps taken if it has been determined that there is a DBA LOCA event in progress. Therefore, should a LOCA occur while the plant is in the MODE 3 drain-down condition, the sprays will be manually initiated by operators as assumed in the DBA LOCA analysis. While in MODE 3, the following observations are made:

a.) Additional radiological decay in the source term occurs due to the time that is required to reach MODE 3 conditions versus the DBA hot full power LOCA initial condition, and

b.) Lower containment leakage rates will exist in the event of a LOCA in MODE 3 due to the lower containment pressure

Therefore, the existing DBA LOCA dose analysis bounds the dose consequences that would result from a LOCA that might occur in this MODE 3 drain-down configuration.

3.7 Steam Line Break with Steam Bypass of Suppression Pool

The concept of the pressure suppression reactor containment is that any steam released from the primary system is condensed by the suppression pool and does not have an opportunity to produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. This arrangement forces steam released from the primary system to be condensed in the suppression pool. Should a leakage path exist between the drywell and containment, the leaking steam would result in pressurization of the containment. To mitigate the consequences of any steam bypassing the suppression pool, a high containment pressure signal (9 psig analytical value) automatically initiates the containment spray system any time after LOCA plus ten minutes. The original design basis assumptions for the allowable bypass calculations are that containment spray is

activated 180 seconds after containment pressure reaches 9 psig or at LOCA plus 13 minutes, whichever occurs later.

When the suppression pool level is increased, the pressure in the drywell required to clear the top vent is also increased. GOTHIC analyses were performed to determine the impact of raising the suppression pool level on this steam bypass capability analysis at the reduced vessel pressure of 235 psig.

The GOTHIC model was benchmarked against the bypass capability analysis discussed in USAR, Section 6.2.1.1.5.4 (Reference 3). The steam bypass analysis described in the USAR predicted a maximum drywell bypass leakage capability of $A/\sqrt{K}$ = 1.68 ft^2 at full-power conditions that would ensure the containment design pressure limit of 15 psig would not be exceeded. A GOTHIC benchmark analysis was performed at the same conditions of the USAR analysis, which included containment spray activation at 180 seconds after containment pressure reaches 9 psig or at LOCA plus 13 minutes, whichever occurs later. A spectrum of break sizes ranging from 0.07 ft^2 to 3.5 ft^2 was analyzed to determine the limiting break size. A break size of 0.5 ft^2 produced the limiting containment pressure response (Attachment 4, Figure 8) that satisfied the containment design pressure limit.

The containment pressure response during a steam bypass event in MODE 3 drain-down conditions at 235 psig was analyzed for a drywell bypass leakage capability of $A/\sqrt{K}$ = 1.68 ft^2. Break sizes ranging from 0.07 ft^2 to 3.5 ft^2 were again evaluated. Containment spray was credited at 180 seconds after containment pressure reached 9 psig or at LOCA plus 13 minutes, whichever occurred later. The initial level in the suppression pool was increased to compensate for the entrapment volumes developed for the MODE 3 drain-down conditions (as described in Section 3.2.2), which considered containment spray collecting in the reactor well pool. The initial suppression pool level in the analysis was placed near the top of the drywell weir wall (near 24 feet 2 inches) as a conservatism in excess of the upper bound analytical limit of 23 feet 8 inches. Figure 13 in Attachment 4 shows a peak containment pressure of 29.66 psia (14.96 psig) for the most limiting break size (0.5 ft^2), which is just below the containment design limit of 29.70 psia (15 psig). This information, as illustrated in Figure 13 of Attachment 4, shows that the peak pressure complies with the containment pressure design limit.

3.8 Miscellaneous Considerations

3.8.1 Potential Drywell Flooding with Inadvertent UCP Dump

To minimize drywell flooding during normal operations and transients, the weir wall height is designed to limit overflow into the drywell. The normal freeboard from the TS HWL is 5 feet 8 inches (top of drywell weir wall is 24 feet 2 inches, HWL is 18 feet 6 inches). An inadvertent dump of the upper pool during any period of operation with a pressurized vessel does not represent, in and of itself, any hazard to the public, the plant operating personnel, or any plant equipment. The drywell weir wall has sufficient freeboard height between the suppression pool surface and the top of the weir wall to

store most of the upper pool makeup volume on top of the normal suppression pool HWL with limited flooding over the weir wall into the drywell under negative drywell pressure conditions. The piping components which would be affected in this event have been analyzed for the flooding affect, as discussed in Perry USAR Section 6.2.7.3.3. This section of the USAR also indicates that this event could not initiate a LOCA.

For MODE 3 drain-down when the reactor well is fully drained, the proposed suppression pool level at the upper analytical limit is 23 feet 8 inches (as described in Section 3.2.2), which corresponds to 5 feet 2 inches above the current TS HWL. At the analytical suppression pool level upper limit, the available freeboard is 6 inches. If the suppression pool water level is at the upper analytical limit and there is an inadvertent UCP dump, water will over flow the weir wall. To minimize the flooding during normal operations or during a hypothetical inadvertent dump of the UCP, a zero to positive differential pressure between the drywell and containment will be administratively maintained during and following the proposed MODE 3 drain-down (until MODE 4). There is a high reliability that the UCP will be dumped to the suppression pool when required, but not dumped inadvertently by spurious signals or operator error.

3.8.2 UCP Dump Time Versus Suppression Pool Pump Time Criterion

During the time when the ECCS pumps are pumping water out of the suppression pool, water level will decrease until it reaches the low-low water level (LLWL). At that level, the dump valves in the SPMU system will open and the water inventory available in the UCP will dump to the suppression pool. Water will flow into the suppression pool from the UCP at the same time water continues to flow out of the suppression pool via the ECCS pumps. This condition is evaluated to verify that the flow of water into the suppression pool from the SPMU system is sufficient to ensure that the water coverage above the horizontal vents is not compromised.

To ensure drywell vent coverage of 2 feet is maintained during 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 pump time is determined by dividing the pumping volume (upper pool makeup volume plus the volume in the suppression pool stored between the LLWL and the minimum top vent coverage) by the total maximum runout flow rate from all five ECCS pumps.

The pumping volume considers the suppression pool makeup volume, which is reduced following reactor well gate installation and reactor well drain. An analysis of the SPMU dump time was performed for operations with the gate installed and with the reactor well pool drained. With gates installed (MODES 1, 2, and 3) and the UCP level at 23 feet 0 inches above RPV flange, the allowable dump time is approximately 12 seconds less than the pump time. With the reactor well pool drained in MODE 3, the allowable dump time is 3 seconds less than the pump time. Therefore, the SPMU "dump time" criterion is met considering the decreased makeup volume available during the proposed operations.

3.8.3 Evaluation of IFTS Blind Flange Removal during the MODE 3 Drain-Down Activity

Gate installation in MODES 1, 2 and 3 will occur at the end of a given operating cycle to facilitate MODE 3 drain-down of the reactor well as the plant is in the process of shutting down for a refueling outage. During this time, the blind flange in the inclined fuel transfer system (IFTS) penetration can be removed, as permitted by TS surveillance requirement (SR) 3.6.1.3.4, Note 4. Note 4 permits the removal of the IFTS blind flange, provided certain other requirements are met and limits its removal to no more than 60 days per cycle while in MODES 1, 2, or 3. This action is taken by plant workers shortly before each refueling outage to accommodate testing, IFTS system maintenance, or implementation of IFTS design modification work prior to the plant refueling outage. License Amendment No. 100, which permitted the blind flange removal, was approved by NRC letter dated February 24, 1999 (Reference 6). Supplemental restrictions were placed on this activity as described in a PNPP LAR submittal letter dated March 14, 2002 (Reference 7) and subsequent License Amendment No. 123, which was approved by NRC letter dated March 7, 2003 (References 8 and 9).

With the IFTS flange removed at power and the upper pool IFTS gate removed, the potential exists to drain the upper containment pools and reduce the inventory available to the SPMU system. The removal of the safety-related IFTS blind flange results in reliance on non-safety equipment to maintain pool level. Installation of the upper pool IFTS gate was therefore required prior to IFTS blind flange removal, in accordance with TS SR 3.6.1.3.4, Note 4. Failure of the IFTS would still result in draining of the fuel transfer pool, although that inventory is not needed for the SPMU system. Although upper pool IFTS gate installation is required, it was identified that failure of the IFTS could still impact the steam dryer storage pool and other upper containment pools since flow paths exist via four 1-inch siphon breaker lines, which connect the fuel transfer pool to each of the upper containment pools. Additional controls were established to maintain suppression pool water level at least 17 feet 11.7 inches and the UCP water level at least 22 feet 9 inches above the reactor flange to address this condition.

A probabilistic safety assessment (PSA) evaluation was performed for Amendment No. 123, which conservatively assumed that the IFTS gate was not installed and that IFTS failure would reduce the SPMU system volume and fail the SPMU function. The estimated increase in core damage frequency (CDF) was determined and was found to be below the guideline value for assuring that the increase in risk associated with the license amendment request was small and consistent with the intent of the Nuclear Regulatory Commission's Safety Goal Policy Statement. The estimated incremental conditional core damage probability (ICCDP) was also determined and was found to be below the guideline value for confirming that a proposed permanent TS change has only a small quantitative impact on plant risk. Based on those results, the NRC staff concluded that the proposed LAR, associated with Amendment No. 123, had an acceptably small impact on CDF.

Regarding this proposed license amendment, loss of water inventory from the fuel transfer pool and possibly the steam dryer storage pool during MODE 3 drain-down

would create an additional entrapment volume in the event containment spray was initiated. As assumed in the MODE 3 drain-down analyses, water level in these pools during MODE 3 drain-down is required to remain at or above elevation 688'-3" (22 feet 8 inches above the RPV flange). This allows containment spray water to be trapped in these pools only between elevation 688'-3" and the top of the partition wall (at elevation 688'-5") that separates the steam dryer storage pool from the reactor well. Any potential draining of water below the assumed 22 feet 8 inch level in either or both pools will create entrapment areas not accounted for in the analyses.

Since IFTS failure can potentially create an additional undesired entrapment volume in the event containment spray is actuated, an assessment was performed to determine if a postulated IFTS failure would have an impact on the analyses and evaluations that were done in support of this proposed amendment. The review concluded that with the employment of additional defense-in-depth measures, failure potentials [related to human error and single point vulnerabilities] would be reduced below a level of credibility that could influence the proposed amendments analyses. These defense-in-depth measures include placement of the IFTS carriage in the upper pool, suspension of IFTS activities, and closing the manual maintenance valve. With these measures in place, failures associated with human error potential due to maintenance, human error potential associated with IFTS drain valve closure, and single component failure potentials are removed. The plant's emergency operating procedures also contain mitigative guidance to maintain suppression pool levels within acceptable limits in the event inventory was lost. This guidance was conservatively not credited in the qualitative evaluation.

The compensatory measures identified above are included in the proposed TS 3.10.9.

4.0 REGULATORY EVALUATION

FirstEnergy Nuclear Operating Company (FENOC) is requesting amendment of Operating License NPF-58 for the Perry Nuclear Power Plant, Unit No. 1 (PNPP) to revise two Technical Specifications (TSs) and create another TS to support upper containment pool (UCP) drain-down in Mode 3 (Hot Shutdown). The proposed amendment would allow installation of the reactor well to steam dryer storage pool gate in the upper containment pool (UCP) in MODES 1, 2, and 3, and permit draining the reactor well pool portion of the UCP while still in MODE 3.

The TSs that would be revised by the proposed amendment are as follows:

- TS 3.6.2.2, "Suppression Pool Water Level"
- TS 3.6.2.4, "Suppression Pool Makeup (SPMU) System"

The TS that would be created is:

- Special Operations TS 3.10.9, "Suppression Pool Makeup - MODE 3 Upper Containment Pool Drain-Down"

4.1 No Significant Hazards Consideration Determination

FENOC 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 amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The changes proposed in the license amendment request specify different water level requirements in the upper containment pool and suppression pool to permit gate installation in MODES 1, 2, and 3, and drain-down of the reactor well in MODE 3. The probability of an accident previously evaluated is unrelated to the water level in these pools, since they are mitigating systems. The operation or failure of a mitigating 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.

   Suppression pool water levels are increased during upper pool gate installation in MODES 1, 2, and 3 and during reactor well drain-down in MODE 3, with a potential for an increased probability of drywell flooding during an inadvertent dump of the upper containment pool. An inadvertent dump of the upper pool during any period of operation with a pressurized vessel does not represent, in and of itself, any significant hazard to the public, the plant operating personnel, or any plant equipment. The piping components which would be affected in this event have been analyzed for the flooding effect, and it has been determined that this event could not initiate a loss of coolant accident (LOCA).

   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 maintained. 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 level for the proposed reactor well drain-down activity 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.

Relative to dose rates on the refuel floor, the resultant dose rates from the reactor in MODES 3 and 4 are the same regardless of a drain-down of the upper pool reactor well. Relative to a low pressure LOCA in MODE 3, the reduced post-LOCA containment pressure and the decay time to reach MODE 3 conditions ensures that post-accident dose consequences are bounded by the design-basis accident LOCA.

Therefore, the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes specify different water level requirements in the upper containment pool and suppression pool to permit gate installation in MODES 1, 2, and 3, and drain-down of the reactor well in MODE 3. These changes do not affect or alter the ability of the suppression pool makeup (SPMU) system to perform its design function. The proposed change in the pool water levels will maintain the design function of mitigating the pressure and temperature increase generated by a LOCA, and will maintain the required drywell vent coverage during post-accident ECCS draw down.

   The altered water levels in the pools do not create a different type of accident than presently evaluated. With the reduced pressure in the reactor coolant system, the GOTHIC computer program simulations demonstrate that the accident responses at defined conditions with the reactor well drained in MODE 3 are bounded by the current design basis accidents.

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

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

   Response: No.

   The proposed changes to the UCP and the suppression pool water levels do not introduce any new setpoints at which protective or mitigating actions are initiated. Current instrument setpoints remain unaltered by this change. Although the water levels are adjusted for the UCP gate installation and the reactor well drain-down activity, the design and functioning of the containment pressure suppression system remains

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 accident results remain within the design values for the structures and equipment.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, FENOC concludes that the proposed amendment does not involve a 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.

4.2 Applicable Regulatory Requirements/Criteria

Appendix A to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," identifies the General Design Criteria (GDC) for overall requirements, protection by multiple fission product barriers, protection and reactivity control systems, fluid systems, reactor containment, and fuel and reactivity control for nuclear power plants. From those criteria, the following are considered for applicability to this LAR.

GDC 4, "Environmental and Dynamic Effects Design Bases," requires that structures, systems, and components important to safety (such as the containment and the suppression pool) shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant-accidents and that they shall be appropriately protected against dynamic effects. NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 6.2.1.1.c, area of review item 3, indicates that this includes suppression pool dynamic effects during a LOCA or following the actuation of one or more reactor coolant system safety/relief valves. The proposed license amendment will raise the level of the suppression pool above the current HWL limit. This will affect the hydrodynamic loads on the containment structure, including the drywell and the suppression pool. The hydrodynamic loads associated with the proposed amendment have been evaluated and have been found to be acceptable, as discussed in Section 3.3.

GDC 16, "Containment Design," requires the containment to be a leak-tight barrier and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The initial water inventory required in the suppression pool and the additional water available from the SPMU system have been evaluated for the proposed UCP gate installation in MODES 1, 2, and 3, and for the proposed drain-down evolution in MODE 3 to ensure that containment design remains in compliance with the requirements of GDC 16. Adequate post-accident vent coverage has been verified for the proposed configurations, as discussed in Section 3.2. Short-term drywell pressure responses for the MSLB and

RSLB events remain below drywell design limits (reference Figures 9 and 10 in Attachment 4). As discussed in Section 3.5, analyses for long term heat sink indicate that the peak long-term suppression pool (SP) temperatures for the MSLB and RSLB events remain below the design temperature limit for the SP (reference Figures 11 and 12 in Attachment 4). As discussed in Section 3.7, steam line break with steam bypass of the SP has been evaluated and the results indicate that peak containment pressure remains below the containment design limit (reference Figure 13 in Attachment 4). As discussed in Section 3.8, other miscellaneous considerations related to containment design have also been evaluated and have been found to be acceptable. The proposed amendment does not compromise containment design.

GDC 38, "Containment Heat Removal," requires that the containment heat removal system remove heat from the reactor containment following a LOCA so that the containment pressure and temperature following a LOCA will be maintained at acceptably low levels. Resultant drywell and containment pressures have been evaluated at the MODE 3 drain-down conditions and have been found to be acceptable (reference Figures 9, 10 and 13 in Attachment 4). Long-term SP temperatures have been evaluated for the MSLB and RSLB events and have been found to remain below the design temperature limit for the SP (reference Figures 11 and 12 in Attachment 4). The proposed amendment does not compromise containment heat removal capability.

GDC 50, "Containment Design Basis," requires that the reactor containment structure and its internal compartments accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. The margin must include conservatism of the calculation model and input parameters. The steam line break with steam bypass of SP event has been evaluated at the MODE 3 drain-down conditions. Analyses were performed using the GOTHIC computer program to determine the impact of raising the SP level on the steam bypass capability analysis at the reduced MODE 3 vessel pressure. Figure 13 in Attachment 4 illustrates that the peak containment pressure for the most limiting break size remains below the containment pressure design limit. The use of the GOTHIC computer program is discussed in Attachment 4.

As discussed above, the proposed changes do not affect conformance with any of the applicable General Design Criteria.

4.3 Precedent

The proposed amendment herein is similar to a license amendment for the Grand Gulf Nuclear Station (GNSS) that was approved by the NRC in a letter dated September 6, 2002 (Reference 10), and a license amendment for the Clinton Power Station (CPS) that was approved by the NRC in a letter dated June 12, 2003 (Reference 11).

Both precedents are similar to this amendment with regard to plant type, intended operations, and analysis method. The past submittals involved the evaluation of a GE BWR/6 with a Mark III containment similar to the PNPP. In both submittals, the utility

requested drain-down of the upper containment pool during MODE 3 for a reactor dome pressure less than or equal to 235 psig (analytical limit). Both submittals used the GOTHIC computer code to perform a plant-specific benchmark of the design basis loss of coolant accidents (LOCAs) for comparison against the licensing basis results, as well as evaluation of these same events at the MODE 3 drain-down conditions.

There are differences between this proposed amendment and the past precedents, which include plant physical characteristics and assumptions made in the associated analyses. Plant physical characteristics include the size of the post-LOCA entrapment volumes, dimensions of the upper containment pool, and dimensions of the suppression pool.

Comparison to Grand Gulf Nuclear Station (GGNS):
PNPP is similar with regard to the post-accident entrapment volumes; however, GGNS has a larger suppression pool and steam separator storage pool. Consequently, the necessary inventory to be added to the GGNS suppression pool to offset the post-accident entrapment volumes during a MODE 3 drain-down is much less in comparison to PNPP. The higher proposed suppression pool levels needed for PNPP during MODE 3 drain-down are also a function of the comparative suppression pool geometries (that is, adding a given amount of water to the larger GGNS suppression pool will raise level less than it will in the PNPP suppression pool).

Comparison to Clinton Power Station (CPS):
Likewise, in comparison to the CPS, the post-accident entrapment volume at CPS is smaller than the post-accident entrapment volume at PNPP. This difference is largely attributable to the drywell pool dimensions, where CPS has a smaller drywell pool. In addition, the suppression pool at CPS is larger than the suppression pool at PNPP. The size of these structures and the difference in entrapment volumes permitted CPS to add less inventory to the suppression pool for MODE 3 drain-down in comparison to the proposed amount at PNPP. Again, the higher proposed suppression pool levels needed for PNPP during MODE 3 drain-down are a function of the comparative suppression pool geometries (that is, adding a given amount of water to the larger CPS suppression pool will raise level less than it will in the PNPP suppression pool).

The proposed amendment herein also differs in the analytical approach in addressing some of the post-accident entrapment volumes.

RPV entrapment volume:
The RPV entrapment volume in the GGNS drain-down analyses considered the RPV region between the normal operating water level and the Level 8 control band elevation. The CPS drain-down analyses defined the RPV entrapment volume between the normal operating water level to just above the bottom of the MSL nozzles. For the PNPP drain-down analyses, the RPV entrapment volume was extended up to the centerline elevation of the MSL nozzles so as to address

expected RPV levels during a large MSLB that discharges water out of the break. For small line breaks, operators are expected to control RPV water level below the Level 8 control band during a post-LOCA event. However, in the event operators declare the Level 8 instrument readings unavailable, PNPP EOPs instruct operators to open the main steam safety relief valves, which dump to suppression pool, and increase ECCS injection to the RPV. Under these conditions, ECCS injection will flow out the main steam S/RVs or out the break on the main steam line, thereby limiting the RPV flooding to near the centerline elevation of the MSL nozzles.

Reactor well entrapment volume:
Unlike CPS and GGNS, PNPP evaluations included the reactor well as an additional entrapment volume for containment sprays. Although containment sprays are not expected to actuate during a design basis event (large MSLB and RSLB), the additional entrapment volume in the reactor well has been accounted for, should PNPP operators choose to manually actuate sprays. Consequently, the PNPP approach need not consider additional operator actions post-accident to maintain necessary water levels in the suppression pool, as was necessary for GGNS. The PNPP approach implements a drain-down curve that considers the reactor well as an increasing entrapment volume as it is drained in MODE 3.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve, (i) a significant hazards consideration, (ii) a significant change in the types of 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 change 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. USAR Appendix 3B, "Containment Loads."

2. USAR Chapter 5, "Reactor Coolant System and Connected Systems."

3. USAR Chapter 6, "Engineered Safety Features."

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

5. NUREG-0887, Supplement No. 8, "Safety Evaluation Report related to the operation of Perry Nuclear Power Plant, Units 1 and 2," January 1986.

6. NRC Letter, "Amendment No. 100 to Facility Operating License No. NPF-58-Perry Nuclear Power Plant, Unit 1 (TAC No. MA3486)," February 24, 1999 (Accession No. ML021840212).

7. PNPP Letter PY-CEI/NRR-2614L, "License Amendment Request Pursuant to 10CFR50.90: Inclined Fuel Transfer System (IFTS)," March 14, 2002 (Accession No. ML020870456).

8. NRC Letter, "Perry Nuclear Power Plant, Unit 1 - Issuance of Amendment [No. 123] (TAC No. MB4694)," dated March 7, 2003 (Accession No. ML030360652).

9. Technical Specification Pages for Amendment No. 123 (Accession No. ML030700167).

10. NRC Letter, "Grand Gulf Nuclear Station, Issuance of Amendment Re: Reactor Cavity Pool Draindown (TAC No. MB4260)," dated September 6, 2002 (Accession No. ML022490416).

11. NRC Letter, "Clinton Power Station, Unit 1 - Issuance of Amendment (TAC No. MB3578)," dated June 12, 2003 (Accession No. ML031290397).

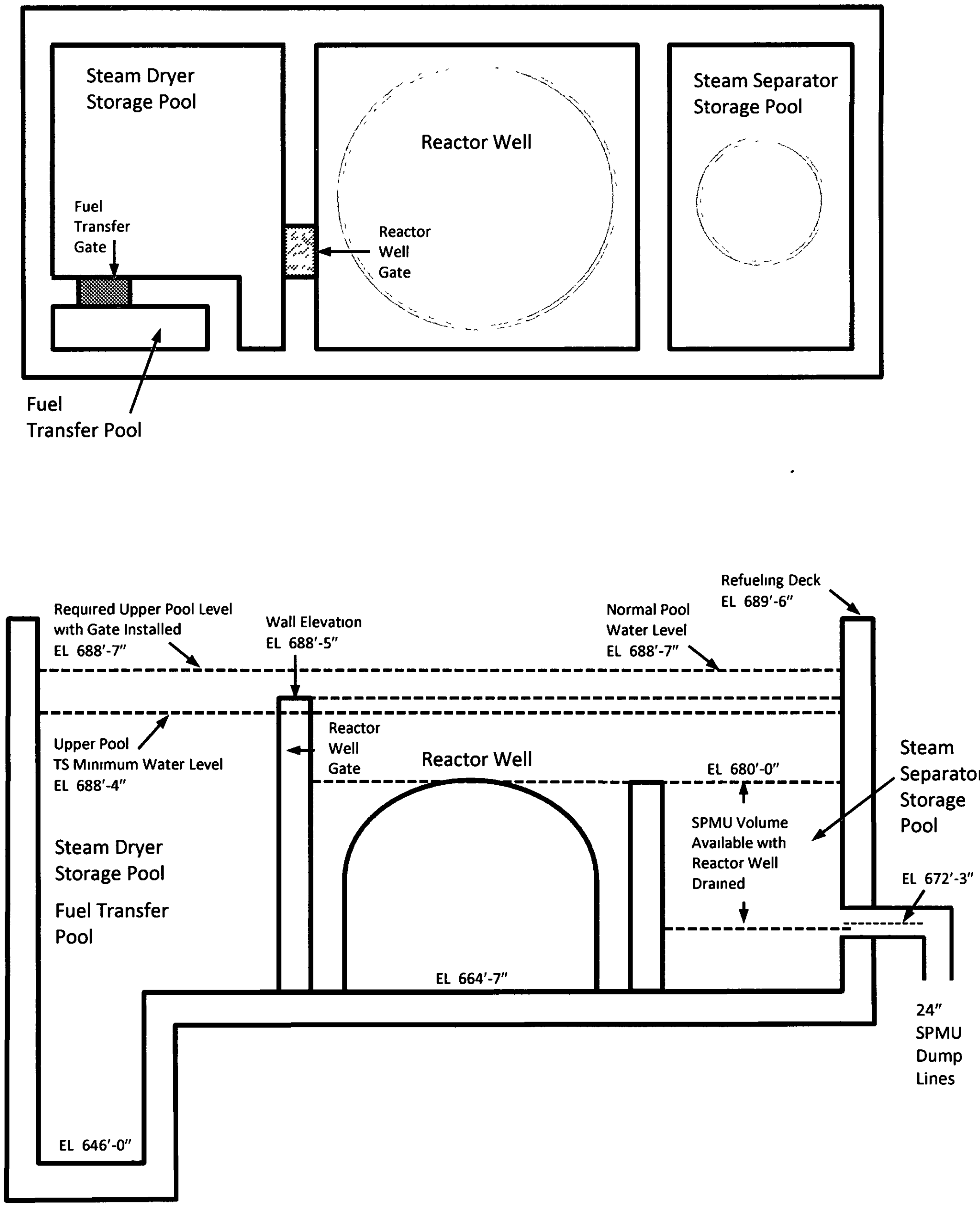


Figure 1: Upper Containment Pool Arrangement, Water Levels and Elevations (Not to Scale)

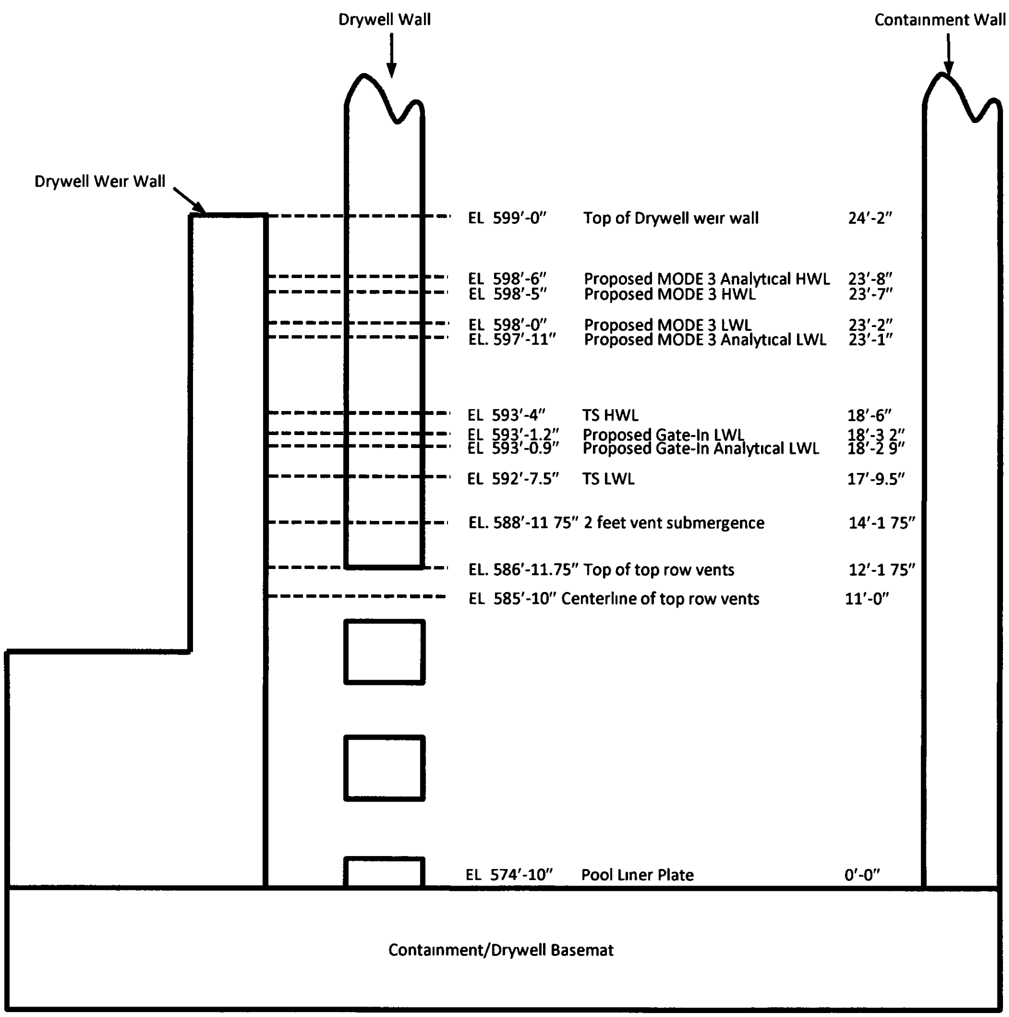


Figure 2: Suppression Pool Water Levels
(Not to Scale)

Attachment 1

Proposed Technical Specification Changes (Mark-Up)
(8 pages follow)

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Corrected suppression pool water level shall be ≥ 17 ft 9.5 inches and ≤ 18 ft 6 inches, when the reactor well to steam dryer storage pool gate is not installed,

OR

Corrected suppression pool water level shall be ≥ 18 ft 3.2 inches and ≤ 18 ft 6 inches, when the reactor well to steam dryer storage pool gate is installed.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool water level not within limits.	A.1 Restore suppression pool water level to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.2.1	Verify suppression pool water level is within limits.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Suppression Pool Makeup (SPMU) System

LCO 3.6.2.4 Two SPMU subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Combined upper containment pool and suppression pool water levels not within limit.	A.1 Restore combined upper containment pool and suppression pool water levels to within limit.	4 hours
B. Upper containment pool water temperature not within limit.	B.1 Restore upper containment pool water temperature to within limit.	24 hours
C. One SPMU subsystem inoperable for reasons other than Condition A or B.	C.1 Restore SPMU subsystem to OPERABLE status.	7 days
D. Required Action and associated Completion Time of not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify upper containment pool water level is: a. ≥ 22 ft 9 inches above the reactor pressure vessel (RPV) flange, when the reactor well to steam dryer storage pool gate is not installed. OR b. ≥ 22 ft 5 inches above the RPV flange, and suppression pool water level ≥ 17 ft 11.7 inches, when the reactor well to steam dryer storage pool gate is not installed. OR c. ≥ 23 ft 0 inches above the RPV flange and the suppression pool water level ≥ 18 ft 3 2 inches, when the reactor well to steam dryer storage pool gate is installed.	24 hours
SR 3.6.2.4.2	Verify upper containment pool water temperature is ≤ 110° 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 positon is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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.c, no work is being performed that has the potential to drain the upper fuel transfer pool, IFTS carriage is located in the upper pool, and IFTS transfer tube shutoff valve 1F42F002 is closed. -- Verify all required upper containment pool gates are in the stored positon 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.	24 months

3.10 SPECIAL OPERATIONS

3.10.9 Suppression Pool Makeup – MODE 3 Upper Containment Pool Drain-Down

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 ≤ 110°F,

b. Suppression pool and upper containment pool water levels are maintained within limits of Figure 3.10.9-1;

c. The steam dryer storage pool and the fuel transfer pool areas of the upper containment pool are maintained at a minimum of 22 ft 8 inches above the reactor pressure vessel (RPV) flange;

d. Reactor steam dome pressure is ≤ 230 PSIG;

e. Reactor has been subcritical ≥ 2 hours;

f. Each SPMU subsystem valve is OPERABLE in accordance with 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;

g. No work is being performed that has the potential to drain the upper fuel transfer pool,

h. IFTS carriage is located in the upper pool;

i. IFTS transfer tube shutoff valve 1F42F002 is closed; and

j. Reactor well to steam dryer storage pool gate is installed.

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

ACTIONS

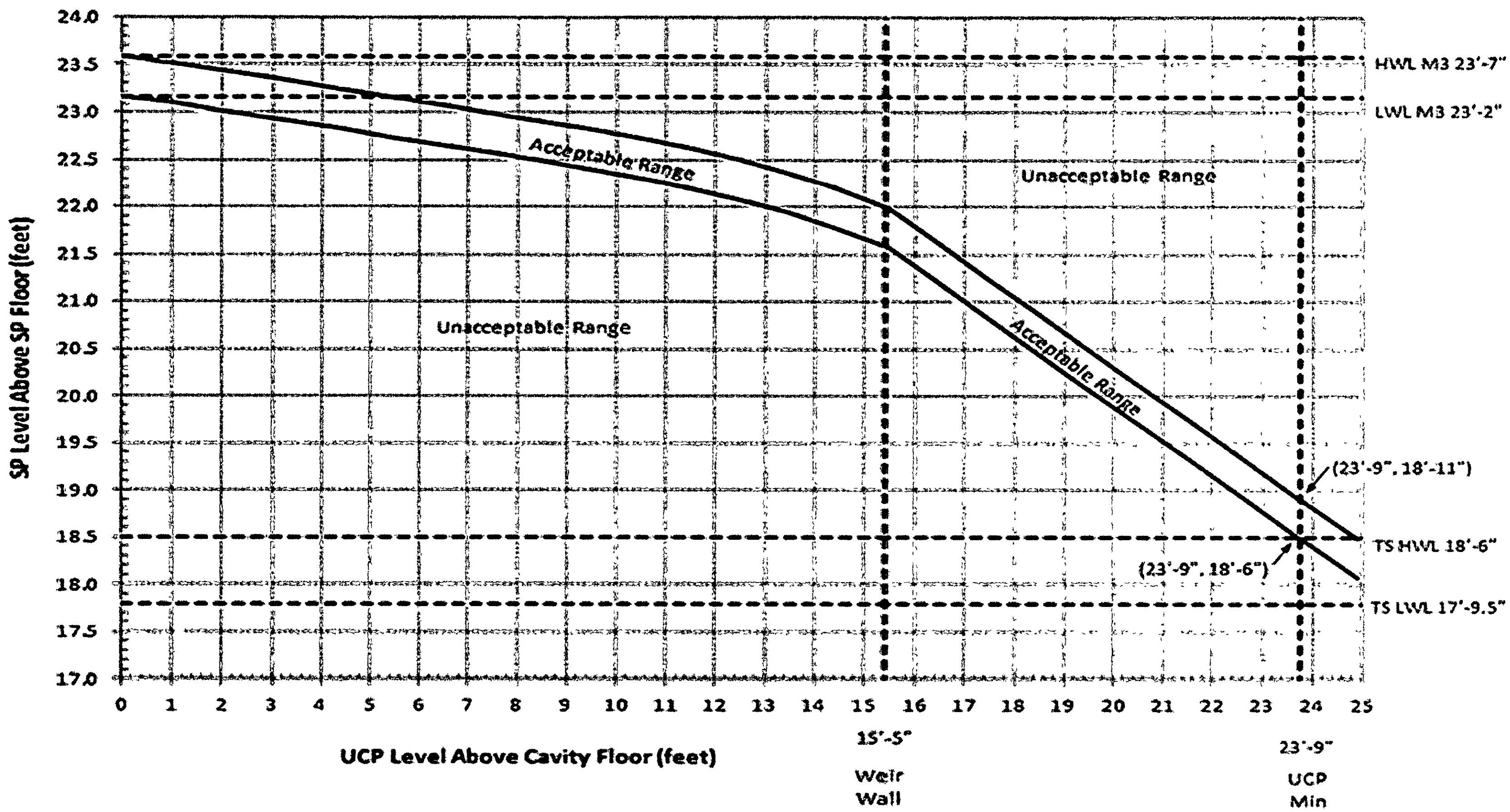
-----NOTE-----
Separate Condition entry is allowed for each requirement of the LCO.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the above requirements not met.	A.1 Suspend draining the upper containment pools. AND A.2 Restore compliance with the requirements of this LCO.	Immediately 4 hours
B. Required Action and Completion Time of Condition A not met	B.1 Restore compliance with suspended MODE 3 LCO requirements.	12 hours
C Required Action and associated Completion Time of Condition B not met	C 1 Be in MODE 4.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.9.1	Verify suppression pool temperature is ≤ 110°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 steam dryer storage pool and the fuel transfer pool areas of the upper containment pool are ≥ 22 ft 8 inches above the RPV flange.	12 hours
SR 3.10.9.5	Verify IFTS carriage is located in the upper pool and IFTS transfer tube shutoff valve 1F42F002 is closed.	12 hours

Figure 3.10.9-1
Upper Containment and Suppression Pool Levels



Note: UCP water level is measured from the reactor well (cavity) floor and not the reactor pressure vessel (RPV) flange.

Attachment 2

Proposed Technical Specification Changes (Retyped)
For Information Only
(8 pages follow)

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Corrected suppression pool water level shall be ≥ 17 ft 9.5 inches and ≤ 18 ft 6 inches, when the reactor well to steam dryer storage pool gate is not installed,

OR

Corrected suppression pool water level shall be ≥ 18 ft 3.2 inches and ≤ 18 ft 6 inches, when the reactor well to steam dryer storage pool gate is installed.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool water level not within limits.	A.1 Restore suppression pool water level to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.2.1	Verify suppression pool water level is within limits.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Suppression Pool Makeup (SPMU) System

LCO 3.6.2.4 Two SPMU subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Combined upper containment pool and suppression pool water levels not within limit.	A.1 Restore combined upper containment pool and suppression pool water levels to within limit.	4 hours
B. Upper containment pool water temperature not within limit.	B.1 Restore upper containment pool water temperature to within limit.	24 hours
C. One SPMU subsystem inoperable for reasons other than Condition A or B.	C.1 Restore SPMU subsystem to OPERABLE status.	7 days
D. Required Action and associated Completion Time of not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify upper containment pool water level is: a. ≥ 22 ft 9 inches above the reactor pressure vessel (RPV) flange, when the reactor well to steam dryer storage pool gate is not installed. <u>OR</u> b. ≥ 22 ft 5 inches above the RPV flange, and suppression pool water level ≥ 17 ft 11.7 inches, when the reactor well to steam dryer storage pool gate is not installed. <u>OR</u> c. ≥ 23 ft 0 inches above the RPV flange and the suppression pool water level ≥ 18 ft 3.2 inches, when the reactor well to steam dryer storage pool gate is installed.	24 hours
SR 3.6.2.4.2	Verify upper containment pool water temperature is ≤ 110° 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 positon is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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.c, no work is being performed that has the potential to drain the upper fuel transfer pool, IFTS carriage is located in the upper pool, and IFTS transfer tube shutoff valve 1F42F002 is closed. ----- Verify all required upper containment pool gates are in the stored positon 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.	24 months

3.10 SPECIAL OPERATIONS

3.10.9 Suppression Pool Makeup – MODE 3 Upper Containment Pool Drain-Down

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 ≤ 110°F;

b. Suppression pool and upper containment pool water levels are maintained within limits of Figure 3.10.9-1;

c. The steam dryer storage pool and the fuel transfer pool areas of the upper containment pool are maintained at a minimum of 22 ft 8 inches above the reactor pressure vessel (RPV) flange;

d. Reactor steam dome pressure is ≤ 230 PSIG;

e. Reactor has been subcritical ≥ 2 hours;

f. Each SPMU subsystem valve is OPERABLE in accordance with 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;

g. No work is being performed that has the potential to drain the upper fuel transfer pool;

h. IFTS carriage is located in the upper pool;

i. IFTS transfer tube shutoff valve 1F42F002 is closed; and

j. Reactor well to steam dryer storage pool gate is installed.

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

ACTIONS

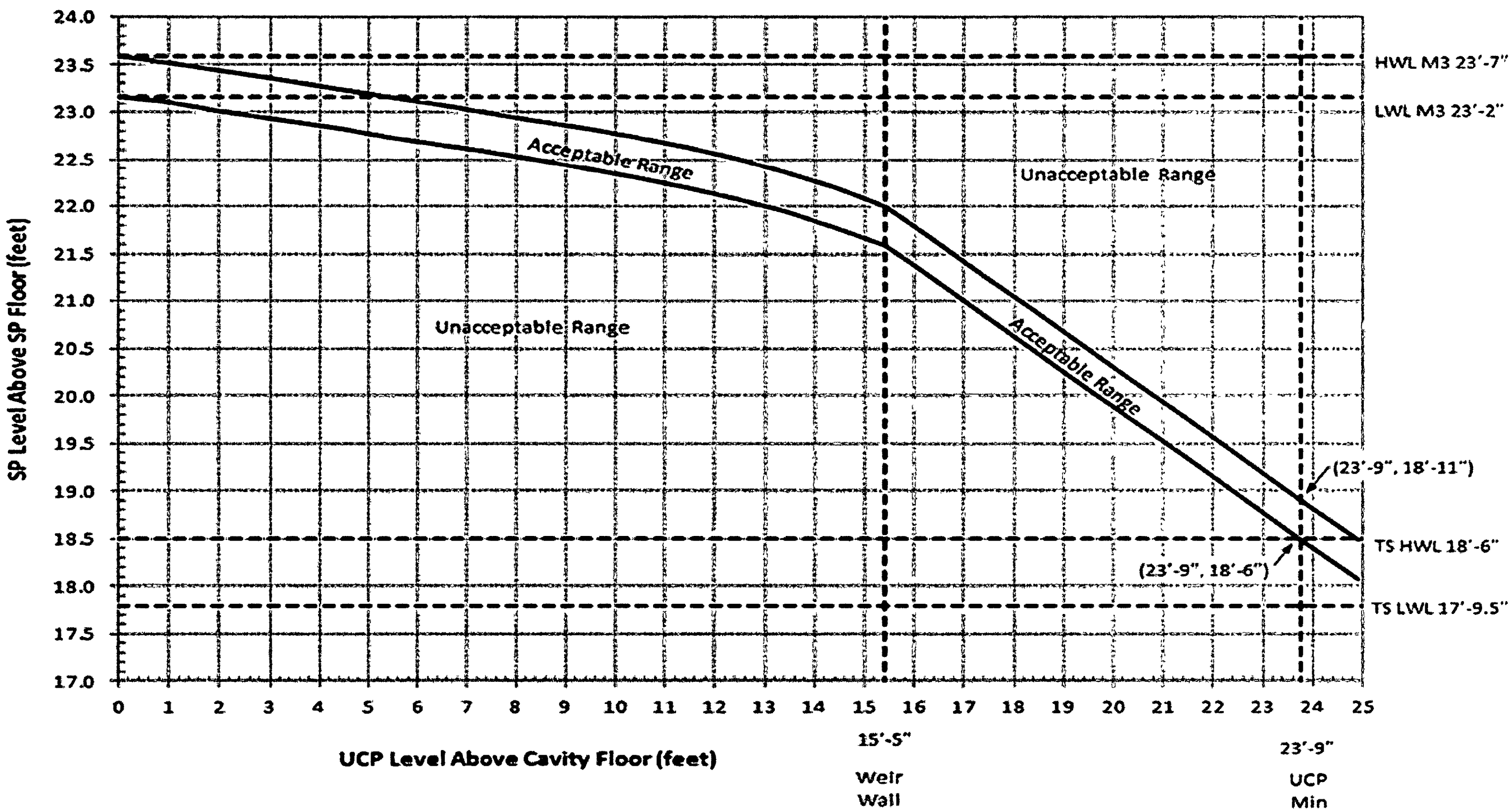
-----NOTE-----
Separate Condition entry is allowed for each requirement of the LCO.

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.9.1	Verify suppression pool temperature is ≤ 110°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 steam dryer storage pool and the fuel transfer pool areas of the upper containment pool are ≥ 22 ft 8 inches above the RPV flange.	12 hours
SR 3.10.9.5	Verify IFTS carriage is located in the upper pool and IFTS transfer tube shutoff valve 1F42F002 is closed.	12 hours

Figure 3.10.9-1
Upper Containment and Suppression Pool Levels



Note: UCP water level is measured from the reactor well (cavity) floor and not the reactor pressure vessel (RPV) flange.

Attachment 3

Proposed Technical Specification Bases Changes (Mark-Up)
For Information Only
(17 pages follow)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

BASES

BACKGROUND

The suppression pool is a concentric open container of water with a stainless steel liner, which is located at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve (S/RV) discharges or from a loss of coolant accident (LOCA). The suppression pool must also condense steam from the Reactor Core Isolation Cooling (RCIC) System turbine exhaust and provides the main emergency water supply source for the reactor vessel.

The high water level limit and the low water level limit (indicated level of 18 ft 6 inches and 17 ft 9.5 inches respectively), are nominal values assuming a zero differential pressure across the drywell wall. These values include the water volume of the containment portion of the pool, the horizontal vents, and the weir annulus (including encroachments).

The suppression pool volume used in the short-term containment LOCA response analyses was 118,131 ft^3, which corresponds to an indicated water level of 18 ft 6 inches with the maximum negative drywell-to-containment differential pressure (-0.5 psid) and primary containment to secondary containment differential pressure (1.0 psid). This volume was used to maximize the negative effect of the suppression pool water volume on the drywell pressure and temperature response.

The suppression pool volume used in the long-term containment LOCA response analyses was 144,292 ft^3, which includes the 32,573 ft^3 makeup volume assumed from the upper containment pool, and corresponds to an indicated water level of 17 ft 9.5 inches with the maximum positive drywell-to-containment differential pressure (2.0 psid). This volume was used to maximize the containment pressure and temperature response results of the long term analyses. The limit on minimum suppression pool water level was set in order to satisfy the analyses for maximum drawdown of the suppression pool.

(continued)

BASES

BACKGROUND (continued)

When the reactor well to steam dryer storage pool gate is installed, the SPMU System available dump volume is reduced by 7472 ft^3. Consequently, the suppression pool level needs to be raised and maintained ≥ 18 ft 3 2 inches to compensate for the loss of volume in the upper containment pool. In addition, the upper containment pool level needs to be maintained ≥ 23 ft 0 inches above the reactor pressure vessel flange in combination with the increased suppression pool minimum water level when the reactor well to steam dryer storage pool storage pool gate is installed (Reference 2).

In order to account for positive drywell-to-containment differential pressures which affect indicated suppression pool water levels (but not volumes), a Suppression Pool Level Adjustment Table is contained in the Plant Data Book. This table lists water level adjustments for various drywell-to-containment differential pressures. The table adjustment factors are used to modify the indicted suppression pool water level to account for the positive drywell-to-containment differential pressures. Negative differential pressures are not required to be adjusted since these differential pressures were directly accounted for in the short-term analyses.

The suppression pool volumes (and corresponding adjusted levels) satisfy criteria or constraints imposed by: (1) maintaining a 2 foot minimum post-LOCA horizontal vent coverage to assure steam condensation/pressure suppression, and to maintain coverage over the RHR A Test Return line, (2) adequate ECCS pump NPSH, (3) adequate depth for vortex prevention, (4) adequate depth for minimum recirculation volume, and (5) minimizing hydrodynamic loads on submerged structures during SRV and horizontal vent steam discharges.

APPLICABLE SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to S/RV discharges. Suppression pool water level must be maintainedwithin the limits specified so that the safety analysis of Reference 1 remains valid. Reference 3 contains an analysis for LOCAs in MODE 3 with reactor pressure equal to 235 psig.

Suppression pool water level satisfies Criteria 2 and 3 of the NRC Final Policy Statement on Technical Specification Improvements (58 FR 39132).

(continued)

BASES

LCO

The limits on suppression pool water level (≥ 17 ft 9.5 inches and ≤ 18 ft 6 inches) are required to assure that the primary containment conditions assumed for the safety analyses are met. Either high or low water level limits were used in the analyses, depending upon which is conservative for a particular calculation. The required suppression pool water level readings depend upon the drywell-to-containment differential pressure. The levels correspond to ≥ 17 ft 9 5 inches and ≤ 18 ft 6 inches for a 0 psid drywell-to-containment differential pressure. Adjusted levels are calculated for positive drywell-to-containment differential pressures to assure a proper suppression pool volume. When the reactor well to steam dryer storage pool gate is installed, the limits on the suppression pool water level are modified to ≥ 18 ft 3.2 inches and ≤ 18 ft 6 inches to assure that the primary containment conditions for the safety analyses are met (Reference 2).

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced because of the pressure and temperature limitations in these MODES. Requirements for suppression pool level in MODE 4 or 5 are addressed in LCO 3.5.2, "ECCS-Shutdown".

ACTIONS

A.1

With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as horizontal vents are covered, RCIC turbine exhaust is covered, and S/RV quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and due to OPERABLE containment sprays. Prompt action to restore the suppression pool water level to within the normal range is prudent, however, to retain the margin to weir wall overflow from an inadvertent upper pool dump and reduce the risks of increased pool swell and dynamic loading. Therefore, continued operation for a limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within specified limits. Also, it takes into account the low probability of an event impacting the suppression pool water level occurring during this interval.

(continued)

BASES

ACTIONS

B.1 and B.2

If suppression pool water level cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2.1

Verification of the suppression pool water level is to ensure that the required limits are satisfied. The 24 hour Frequency of this SR was developed considering operating experience related to water level variations during the applicable MODES and to assessing the proximity to the specified LCO level limits. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool water level condition.

REFERENCES

1. USAR, Section 6.2.
2. Numerical Applications Calculation, NAI-1863-002, Rev. 0, "Perry Nuclear Power Plant UCP Gate Installation Calculation" (Perry Calculation G43-009).
3. Numerical Applications Calculation NAI-1863-001, Rev. 0, "Perry Nuclear Power Plant Early Drain Down in MODE 3" (Perry Calculation 2.2.1.10).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 4 Suppression Pool Makeup (SPMU) System

BASES

BACKGROUND

The function of the SPMU System is to transfer water from the upper containment pool to the suppression pool after a loss of coolant accident (LOCA). For a LOCA, with Emergency Core Cooling System injection from the suppression pool, a large volume of water can be held up in the drywell behind the weir wall This holdup can significantly lower suppression pool water level. The water transfer from the SPMU System ensures a post LOCA suppression pool vent coverage of ≥ 2 ft above the top of the horizontal vents so that long term steam condensation is maintained. The additional makeup water is used as part of the long term suppression pool heat sink. The post LOCA delayed transfer of this water to the suppression pool provides an initially low vent submergence, which results in lower drywell pressure loading and lower pool dynamic loading during a Design Basis Accident (DBA) LOCA as compared to higher vent submergence. The sizing of the residual heat removal heat exchanger takes credit for the additional SPMU System water mass in the calculation of the post LOCA peak containment pressure and suppression pool temperature.

The required water dump volume from the upper containment pool is equal to the difference between the total post LOCA drawdown volume and the assumed volume loss from the suppression pool. The total drawdown volume is the volume of suppression pool water that can be entrapped outside of the suppression pool following a LOCA. The post LOCA entrapment volumes casing suppression pool level drawdown include:

a. The free volume inside and below the top of the drywell weir wall;

b. 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;

c. 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; and

(continued)

BACKGROUND (continued)

d. Allowances for primary containment spray holdup on equipment and structural surfaces.

Therefore as long as the total volume of the suppression pool and the upper containment pool meets the minimum required total water volume assumed in the analyses, the water volume of the upper containment pool can be varied.

The SPMU System consists of two redundant subsystems, each capable of dumping the makeup volume from the upper containment pool to the suppression pool by gravity flow. Each dump line includes two normally closed valves in series. The upper pool is dumped automatically on a suppression pool water level Low-Low signal (with a LOCA signal permissive) or on the basis of a timer following a LOCA signal alone to ensure that the makeup volume is available as part of the long term energy sink for small breaks that might not cause dump on a suppression pool water level Low-Low signal. A 30 minute timer was chosen, since the initial suppression pool mass is adequate for any sequence of vessel blowdown energy and decay heat up to at least 30 minutes.

Although the minimum freeboard distance above the suppression pool high water level limit of LCO 3.6.2.2, "Suppression Pool Water Level," to the top of the weir wall is adequate to preclude flooding of the drywell, a LOCA permissive signal is used to prevent an erroneous suppression pool level signal from causing a pool dump. In addition, the SPMU System mode switch may be keylocked in the "OFF" position to ensure that an inadvertent pool dump will not occur. Inadvertent actuation of the SPMU System during MODE 4 or 5 could create a radiation hazard to plant personnel due to a loss of shield water from the upper pool if irradiated fuel were in an elevated position.

APPLICABLE SAFETY ANALYSES

Analyses used to predict suppression pool temperature following large and small break LOCAs, which are the applicable DBAs for the SPMU System, are contained in References 1 and 2. During these events, the SPMU System is relied upon to dump upper containment 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. The analysis assumes an SPMU System dump volume of 32,573 ft^3 at a temperature of 110°F, with a total water

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

volume of 144,292 ft^3 in the upper containment pool and the suppression pool. Reference 4 contains an analysis for LOCAs in MODE 3 with reactor pressure equal to 235 psig.

The SPMU System satisfies Criterion 3 of the NRC Final Policy Statement on Technical Specification Improvements (58 FR 39132).

LCO

During a DBA, a minimum of one SPMU subsystem is required to maintain peak suppression pool water temperature below the design limits (Ref. 1). To ensure that these requirements are met, two SPMU subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE, assuming the worst case single active failure. The SPMU System is OPERABLE when the upper containment pool water temperature is ≤ 110°F, the piping is intact, and the system valves are OPERABLE. Additionally, the combined water levels of the upper containment pool and the suppression pool must be within limits. When the suppression pool level is maintained 2.2 inches greater than required by LCO 3.6.2.2, "Suppression Pool Water Level", the allowed upper containment pool water level limit is reduced to 22 ft 5 inches. Furthermore, when the reactor well to steam dryer storage pool gate is installed, the allowed upper containment pool water level limit must be maintained ≥ 23 ft 0 inches above the RPV flange, and the suppression pool water level must be increased and maintained at ≥ 18 ft 3.2 inches as per LCO 3.6.2.2 "Suppression Pool Water Level," (Reference 3).

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause heatup and pressurization of the primary containment. In MODES 4 or 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SPMU System OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

When the combined water level of the upper containment pool and suppression pool is not within limits, it is inadequate to ensure that the suppression pool heat sink capability matches the safety analysis assumptions. A sufficient quantity of water is necessary to ensure long term energy sink capabilities of the suppression pool and maintain water coverage over the uppermost horizontal vents. Loss of water volume has a relatively large impact on heat sink capability. Therefore, the combined water level of the upper containment pool and suppression pool must be restored to within limit within 4 hours.

(continued)

BASES

ACTIONS (continued)

A.1

The 4 hour Completion Time is sufficient to provide makeup water to either the suppression pool or the upper containment pool to restore level within specified limit. Also, it takes into account the low probability of an event occurring that would require the SPMU System.

B.1

When upper containment pool water temperature is > 110°F, the heat absorption capacity is inadequate to ensure that the suppression pool heat sink capability matches the safety analysis assumptions. Increased temperature has a relatively smaller impact on heat sink capability. Therefore, the upper containment pool water temperature must be restored to within limit within 24 hours. The 24 hour Completion Time is sufficient to restore the upper containment pool to within the specified temperature limit. It also takes into account the low probability of an event occurring that would require the SPMU System.

C.1

With one SPMU subsystem inoperable for reasons other than Condition A or B, the inoperable subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is acceptable in light of the redundant SPMU System capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

D.1 and D.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1

The upper containment pool water level and, if applicable, the suppression pool water level, is regularly monitored to ensure that the required limits are satisfied. The 24 hour Frequency of this SR was developed considering operating experience related to water level variations during the applicable MODES and considering the low probability of a DBA occurring between surveillances. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to abnormal water level conditions. Reference 3 contains the basis for the required water level in the upper containment pool when the reactor well to steam dryer storage pool gate is installed

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 positon 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 positon would affect only a single subsystem. This Frequency has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 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 steam dryer storage/reactor well pool gate must be removed (or placed in its stored position) to allow communication between the various pool sections. The Surveillance is modified by a Note that allows installation of the steam dryer storage pool to reactor well gate if upper pool level is maintained per SR 3.6.2.4.1 c. Additional restrictions are imposed on the IFTS system to prevent accidental draining of the fuel transfer pool that could detrimentally effect assumptions made within the design basis analyses by creating additional entrapment volume areas for containment sprays (Reference 5). The fuel transfer pool gate may be in place, removed, or placed in its stored position, since the volume of water in the fuel transfer pool is not required for SPMU. 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 verifies 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. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.6 overlaps this SR to provide complete testing of the safety function. The 24 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. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

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.

(continued)

BASES (continued)

REFERENCES

1. USAR, Section 6.2.
2. USAR, Chapter 15.
3. Numerical Applications Calculation, NAI-1863-002, Rev. 0, "Perry Nuclear Power Plant UCP Gate Installation Calculation" (Perry Calculation G43-009).
4. Numerical Applications Calculation NAI-1863-001, Rev. 0, "Perry Nuclear Power Plant Early Drain down in MODE 3" (Perry Calculation 2.2 1.10).
5. PRA Applications Analysis/Assessment Sequence No. PRA-PY1-15-003-R00, Rev. 0 "PRA Assessment of License Amendment Request for Drain Down of the Reactor Cavity Pool While in MODE 3". (Perry TAF 082015).

B 3.6 CONTAINMENT SYSTEMS

B 3.10.9 Suppression Pool Makeup System – MODE 3 Upper Containment Pool Drain-Down

BASES

BACKGROUND	Maintaining the SPMU inventory in the upper containment pools until MODE 4 delays completion of outage work in a timely manner. The purpose of the Special Operations LCO is to allow the upper containment pool to be drained below its normal level in MODE 3 such that certain activities can proceed prior to reaching MODE 4 These activities include installation of the gate between the reactor well and the steam dryer storage pool, and completely draining the reactor well.
APPLICABLE SAFETY ANALYSIS	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 (Reference 2). Supporting analyses differ from those for TS 3.6.2 2 and TS 3.6.2.4 in that a portion of the SPMU volume is assumed to have been transferred to the suppression pool with the remainder available from the steam separator storage pool portion of the upper containment pool. Complete draining of the reactor well will eliminate all of the SPMU system volume in the upper containment pool except for the water volume in the steam separator storage pool below the top of the weir wall that separates the steam separator storage pool from the reactor well. Accordingly, additional hold-up volumes in the reactor well, reactor well drain lines, and a small portion at the top of the steam dryer storage and fuel transfer pools, are considered, which are offset by increases in initial upper containment pool and suppression pool levels prior to reactor well drain-down in MODE 3. The event analyses demonstrate that the containment spray function of RHR is not required following a design basis loss of coolant accident (LOCA) to protect the containment given the reduced temperature and pressure stipulated by the LCO. The analysis results demonstrate that the containment pressure increase following a design basis LOCA in MODE 3 will not be sufficient to result in the auto-initiation of containment spray. However, operators are permitted to take action to manually actuate containment sprays, if warranted, to mitigate containment overpressure and control offsite/control room dose In addition to the design basis analyses, drywell bypass capability analyses (Reference 2) indicate that containment pressure exceeds the containment spray auto-initiation setpoint. Steam bypass leakage and the associated capability analyses are discussed in Reference 2.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

For the most limiting break bypass leakage capability analysis, the containment pressure design basis limit is not exceeded. For the design basis accident (DBA) LOCAs and the steam bypass events, the SPMU system design basis function to maintain post-accident drywell vent coverage is ensured by consideration of all potential post-accident entrapment volumes identified in the design-basis. In addition to the design-basis entrapment volumes, the reactor well, reactor well drain lines, a small portion of the steam dryer storage and IFTS pools are included as potential entrapment volumes for the collection of containment spray when the reactor well is drained in MODE 3.

The containment loads evaluation performed for this special operation including the elevated suppression pool water level 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 and evaluations demonstrated 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 and evaluations supporting the low pressure LCO in MODE 3 are contained in References 1, 2, and 3. During these events, the SPMU system is relied upon to dump the steam separator storage pool water to maintain at least 2 feet of drywell horizontal vent coverage and to provide 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 NRC Policy Statement apply. However, when draining the upper containment pool while in MODE 3, the ACTIONS of the Special Operations LCO shall be met. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying the 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 Special Operations LCO is optional. Operations 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. When draining the upper containment pool while in MODE 3, the ACTIONS of the Special Operations LCO shall be met.

(continued)

BASES

LCO (continued)

Compliance with the Figure 3.10.9-1 level requirements ensures that there is sufficient overlap with the requirements of LCO 3.6.2.2 and 3.6.2.4 such that the combined water volume in containment during the transition to a drained reactor well fulfills the containment water inventory requirements assumed in the analysis. Once the level of the weir wall separating the reactor well from the steam separator storage pool is reached, Figure 3.10.9-1 only applies to the reactor well.

Maintaining the water level in the steam dryer storage pool and in the fuel transfer pool ensures that containment spray water hold-up inside containment is minimized consistent with the supporting analysis.

The reactor subcritical time, suppression pool average temperature, upper containment pool temperature, and reactor steam dome pressure are assumptions of the supporting analyses.

Gate installation in MODES 1, 2 and 3 will occur at the end of a given operating cycle to facilitate early MODE 3 drain down of the reactor well as the plant is in the process of shutting down for the associated refueling outage. During this time, the blind flange in the inclined fuel transfer system (IFTS) penetration can be removed, as permitted by Note 4 in surveillance requirement (SR) 3.6.1.3.4. With the IFTS flange removed at power and the upper pool IFTS gate removed, the potential exists to drain the upper containment pools and reduce the inventory available to the SPMU system. Although installation of the IFTS gate limits potential inventory loss to the IFTS pool, loss of water inventory from the IFTS pool and possibly the steam dryer storage pool during MODE 3 drain-down would create additional undesired entrapment volume(s) in the event containment spray was initiated. Compensatory measures have been identified to reduce any failure potentials below the level of credibility that could influence the supporting MODE 3 drain-down analyses. These compensatory measures include placement of the IFTS carriage in the upper pool, suspension of IFTS activities, and closure of IFTS transfer tube shutoff valve 1F42F002 (Reference 4).

Entry in MODE 4 operation does not require the use of this Special Operations LCO or its ACTIONS.

APPLICABILITY

The MODE 3 requirements stated elsewhere in TS may only be modified by this LCO to allow early drain-down 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.

(continued)

BASES

ACTIONS

A Note has been provided to modify the ACTIONS related to drain-down of the upper containment pools in 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 in the Condition unless specifically stated. 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 in 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.

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 initiates 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 B.1 is an alternative Required Action that can be taken instead of Required Action A.1 and A.2 to restore compliance with the normal MODE 3 requirements, thereby exiting this Special Operations LCOs Applicability. The allowed Completion Time allows sufficient time to re-establish compliance with the appropriate Technical Specification.

C.1

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

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 judgment 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 steam dryer storage pool and fuel transfer 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 3.10.9.5

Verification of IFTS carriage located in the upper pool and that IFTS transfer tube shutoff valve 1F42F002 is closed 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

REFERENCES

1. Numerical Applications Calculation, NAI-1863-002, Rev 0, "Perry Nuclear Power Plant UCP Gate Installation Calculation" (Perry Calculation G43-009).

2. Numerical Applications Calculation NAI-1863-001. Rev. 0, "Perry Nuclear Power Plant Early Drain Down in MODE 3" (Perry Calculation 2.2.1.10).

3. Numerical Applications Report NAI-1863-003, Rev.0, "Perry Draindown Project – Dose Disposition Language", retained in Perry Technical Assignment File (TAF) 082015.

4. PRA Applications Analysis/Assessment Sequence No. PRA-PY1-15-003-R00, Rev. 0, "PRA Assessment of License Amendment Request for Drain Down of the Reactor Cavity Pool While in MODE 3".

Attachment 4

Information Related to the GOTHIC Computer Code

1.0 GOTHIC PROGRAM CODE

The analysis supporting the proposed Technical Specification changes was 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. 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 other nuclear power plants.

For the PNPP analysis, plant-specific benchmarks were performed for the current licensing basis containment analyses described in Section 6.2 of the Updated Safety Analysis Report (USAR). The benchmarking showed generally conservative correlation of the GOTHIC output data relative to the current USAR data generated by General Electric (GE) codes.

The GOTHIC models use a nodal diagram similar to the representative modal diagram shown in Figure 1. In the noding diagram, control volumes are represented as yellow rectangles.

Control volumes in the GOTHIC code contain the system mass and energy. Control volumes are thermal-hydraulically connected by flow paths, which are illustrated as green lines in Figure 1. Figure 1 depicts the key flow paths that were used to analyze the design-basis events including the main steam line break (MSLB), the recirculation suction line break (RSLB), and the main steam line break with steam bypass (SBYP) event that considers steam leakage between the drywell and the containment.

Values of key input parameters used in the GOTHIC models are provided in the tables in this attachment. All of the figures in this attachment are provided in color for clarity of the information.

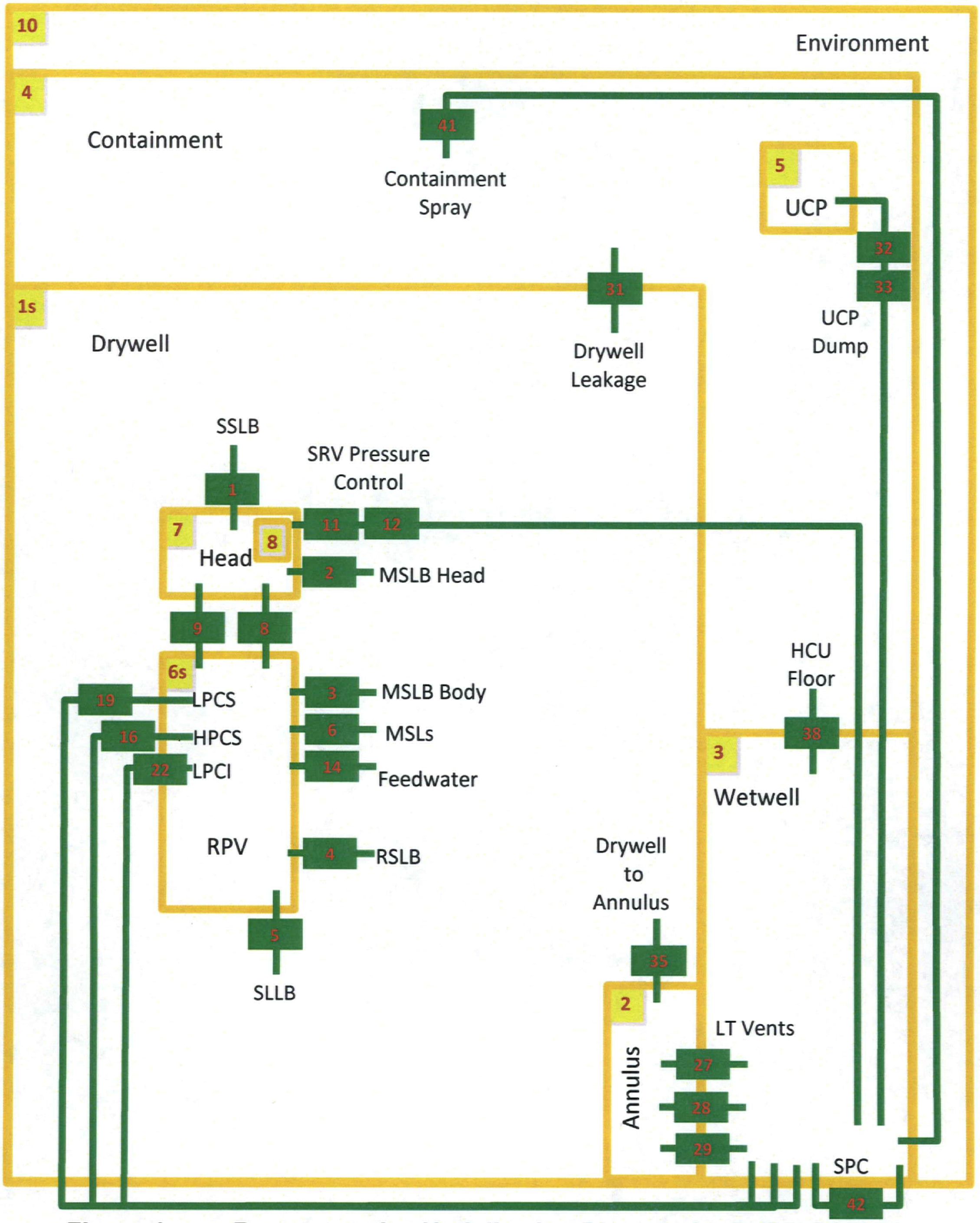


Figure 1: **Representative Nodalization Diagram for PNPP GOTHIC Drain-Down Models**

2.0 GOTHIC PARAMETERS

2.1 Reactor Coolant System (RCS) Initial Conditions

The RCS initial conditions for the benchmark analyses were developed from data listed in USAR Table 6.2-5 (Reference 1) and other PNPP references. The inputs are summarized in Table 1.

Table 1: RCS Initial Conditions for the Benchmark Analyses

Parameter	Value Use in the GOTHIC Benchmark Analysis
Reactor Power Level (MWt)	3833 MWt (102% of rated)
Average Coolant Pressure (psia)	1060
Average Coolant Temperature (°F)	551.8
Total Reactor Coolant Volume in Model (ft^3)	20,901.2
Volume of Liquid in RPV in Model (ft^3)	11,458
Volume of Steam in RPV in Model (ft^3)	9443.2

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 steam lines to the first MSIV.

2.2 Containment Initial Conditions

The containment initial conditions and physical parameters for the benchmark analyses were developed from data listed in USAR Table 6.2-1 (Reference 2), USAR Table 6.2-5 (Reference 1), and other PNPP references. The inputs are summarized in Table 2. The values are generally consistent with the USAR values for the benchmark cases. Minor differences between GOTHIC model nodalization and the original licensing basis methods and codes used for various events required that inputs be varied from these values for specific cases.

Table 2: Initial Containment Parameters

Parameter	Short-term Analysis	Long-term Analysis	Steam Bypass Analysis
Both Benchmark and MODE 3 Drain-Down Analyses ***(Note 1)***			
Drywell pressure, psig	0.5	2.0	0.0
Containment pressure, psig	1.0	0.0	0.0
Drywell Air temperature, °F	135	145	135
Containment Air temperature, °F	95	104	90
Drywell Relative humidity, %RH	40	40	40
Containment Relative humidity, %RH	100	50	60
Upper Pool water temperature, °F	110	110	110
Drywell Free Volume, ft^3	262,191	265,469	265,469
Annulus Free Volume, ft^3	14,058	14,058	14,058
Wetwell Free Volume, ft^3	260,493	260,493	260,493
Containment Free Volume, ft^3	996,820	996,820	996,820
Suppression Pool surface (LWL-HWL), ft^2	5910.8	5910.8	5910.8
Benchmark Analyses ***(Note 2)***			
Suppression Pool temperature, °F	95	95	90
Upper Pool makeup volume, ft^3	32,573	32,573	32,573
Suppression Pool water volume, ft^3 Annulus Water Volume, ft^3	HWL: 106,736 HWL: 11,395	LWL: 100,667 LWL: 11,052	HWL: 106,736 HWL: 11,395
Suppression Pool depth (nominal), ft Annulus Pool depth (nominal), ft	HWL: 18′-6″ HWL: 18′-6″	LWL: 17′-9.5″ LWL: 17′-9.5″	HWL: 18′-6″ HWL: 18′-6″
MODE 3 Drain-Down Analyses ***(Note 3)***			
Suppression Pool temperature, °F	95	110	90
Upper Pool makeup volume, ft^3	<6412	<6412	<6412
Suppression Pool water volume, ft^3 Annulus Water Volume, ft^3	HWL: 138,228 HWL: 14,040	LWL: 100,667 LWL: 11,052	HWL: 138,228 HWL: 14,040
Suppression Pool depth (nominal), ft Annulus Pool depth (nominal), ft	HWL: >23′-8″ HWL: >23′-8″	LWL: 17′-9.5″ LWL: 17′-9.5″	HWL: >23′-8″ HWL: >23′-8″

NOTE 1: The benchmark and MODE 3 drain-down analyses are evaluated at three types of event initial conditions. The event initial conditions are selected so as to exacerbate the accident response in evaluating the figures of merit for a particular event. The short-term event analyses, simulated for 30 seconds, are evaluated to assess drywell and containment overpressure. The long-term event analyses, simulated for greater than 12 hours, are evaluated to assess suppression pool temperatures. The steam bypass capability analyses, simulated for 8 hours, are evaluated to assess the drywell and containment overpressure.

NOTE 2: The parameters listed under "Benchmark Analyses" are specific to the GOTHIC benchmark analyses. The benchmark cases are evaluated consistent with the design-basis events. The benchmark cases and the MODE 3 event analyses differ on the conditions assumed in the upper containment pool and the suppression pool. The nominal pool depths represent the target levels for conditions when the initial drywell to wetwell pressure differential is zero. For the events with a nonzero pressure differential, the target pool volumes are maintained with the pool depths adjusted accordingly for fluid static head.

NOTE 3: The parameters listed under "MODE 3 Drain-Down Analyses" are specific to the GOTHIC MODE 3 drain-down cases. The benchmark cases and the MODE 3 event analyses differ on the conditions assumed in the upper containment pool and the suppression pool. The MODE 3 cases are evaluated assuming the reactor cavity [well] has been drained with limited makeup inventory in the steam separator storage pool.

For the upper pool makeup volume, a slightly conservative (smaller) makeup volume of 6200 ft^3 was used in the GOTHIC analysis to bound target conditions. For the suppression pool and annulus pool depths, a slightly conservative (deeper) pool depth of 24 feet was used in the GOTHIC short-term and steam bypass event analyses to bound target conditions. An annulus pool depth of 24 feet places the water level near the top of the drywell weir wall (near 24 feet 2 inches). The nominal pool depths in the table represent the target levels for conditions when the initial drywell to wetwell pressure differential is zero. For the events with a nonzero pressure differential, the target pool volumes are maintained with the pool depths adjusted accordingly for fluid static head.

2.3 Decay Heat

The decay heat used for the short-term analyses (that is, 0 to 30 seconds) is based on the American Nuclear Society (ANS) 5.0, "Decay Energy Release Rates Following Shutdown of Uranium Fuel Thermal Reactors," with a 20 percent adder. The decay heat used for the long-term benchmark analyses (that is, beyond 30 seconds) is based on the ANS 5.1 1979, "Decay Heat Power in Light Water Reactors," with 2 sigma uncertainty adders. The decay heat used for the long-term MODE 3 drain-down analyses (that is, beyond 30 seconds) is based on the ANS 5.1 1979, "Decay Heat Power in Light Water Reactors," with 2 sigma uncertainty adders and accounts for additional actinides and activation products.

All decay heat functions include the contributions from a fission coastdown, decay heat, fuel relaxation, and metal-water reactions. The decay heat functions are normalized to 3833 MWt, such that the uprated power conditions are enveloped. The low pressure MODE 3 cases assume the reactor has been shutdown for two hours.

2.4 Available Containment Heat Sinks

The available containment heat sinks used in the benchmark cases are unchanged from the design-basis values provided by GE. Containment heat sinks used in the steam bypass event analyses are consistent with those heat sinks discussed in USAR, Section 6.2.1.1.5, and shown in USAR Table 6.2-10 (Reference 3).

3.0 BENCHMARK RESULTS

Key results of the GOTHIC benchmarks against design-bases results provided by GE are presented in Figure 2 through Figure 8. Figure 2 and Figure 3 represent the total break energy from the MSLB and RSLB, respectively. These figures indicate that the energy released from the vessel calculated by GOTHIC compares very well, albeit slightly conservative, with the design-basis event analyses.

Figure 4 and Figure 5 present the short-term drywell and wetwell pressure for the MSLB and RSLB, respectively. The MSLB pressure results indicate that GOTHIC produces a conservative response for the drywell overpressure, although the wetwell pressure compares well with the design-basis event analyses. Likewise, the RSLB pressure results demonstrate that GOTHIC compares well with the design-basis results, although slightly conservative.

Figure 6 and Figure 7 present the long-term suppression pool temperature after a MSLB and RSLB, respectively. The GOTHIC results compare very well with results from the design-basis events.

Figure 8 presents the long-term containment pressure response during a 0.5 ft^2 steam line break evaluated at the maximum permissible drywell steam bypass leakage of 1.68 ft^2 ($A/\sqrt{K}$). The 0.5 ft^2 break size represents the most limiting break size with regard to the containment overpressure based on an evaluated break spectrum ranging from 0.07 ft^2 to 3.5 ft^2. The peak containment pressure response remains below the containment design limit of 15 psig. The GOTHIC results are in general very similar to the USAR containment pressure response illustrated in USAR Figure 6.2-25 (Reference 4), which is evaluated at $A/\sqrt{K}$ equal to 1.00 ft^2.

4.0 MODE 3 RESULTS WITH RPV PRESSURE AT 235 PSIG

Key results from the GOTHIC cases supporting the proposed Technical Specification changes are shown in Figure 9 through Figure 13. Figure 9 and Figure 10 show the short-term drywell and wetwell pressure responses for the MSLB and RSLB event analyses, respectively. Both event analyses are evaluated with a RPV starting pressure of 235 psig, the reactor shutdown for 2 hours, and the suppression pool level starting above the proposed high water level analytical upper limit (that is, greater than 23 feet 8 inches above bottom of suppression pool). The peak drywell pressures are approximately 28 psia and 27 psia for the MSLB and RSLB events, respectively. The

MODE 3 results show at least 10 psi margin to the drywell overpressure results observed in the design-basis event analyses.

Figure 11 and Figure 12 present the long-term suppression pool temperatures for the MSLB and RSLB event analyses, respectively. Both event analyses are evaluated with a RPV starting pressure of 235 psig, the suppression pool level starting at the current TS low water level limit (17 feet 9.5 inches above bottom of suppression pool), the suppression pool temperature starting at the TS maximum permissible limit of 110 °F, and the UCP temperature at 110 °F. The peak suppression pool temperatures are approximately 167 °F and 173 °F for the MSLB and RSLB events, respectively. The MODE 3 results show at least 12 °F margin to the maximum allowable suppression pool temperature of 185 °F.

Figure 13 presents the long-term containment pressure response during a 0.5 ft^2 steam line break evaluated at the maximum permissible drywell steam bypass leakage of 1.68 ft^2 ($A/\sqrt{K}$). The 0.5 ft^2 break size represents the most limiting break size with regard to the containment overpressure. A break spectrum ranging from 0.07 ft^2 to 3.5 ft^2 was considered. The steam bypass event analysis is evaluated with a RPV starting pressure of 235 psig, and the suppression pool level starting at 24 feet, which is slightly above the proposed high water level analytical upper limit of 23 feet 8 inches. The peak containment pressure is 14.96 psig (design limit of 15 psig), which is mitigated by containment sprays activated at 13 minutes (780 seconds) following the break.

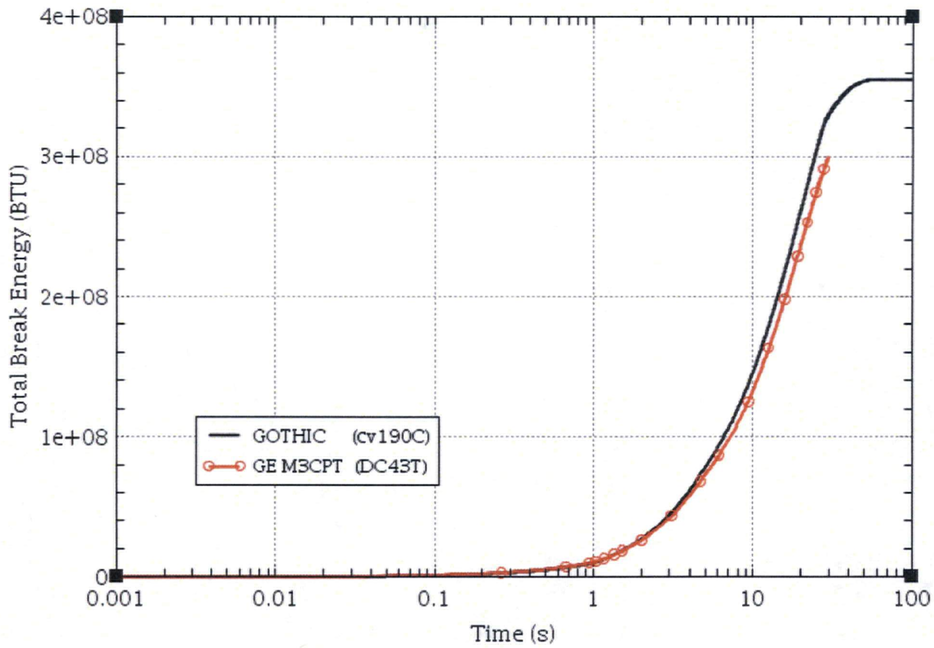


Figure 2: Main Steam Line Break – ST Energy Comparison

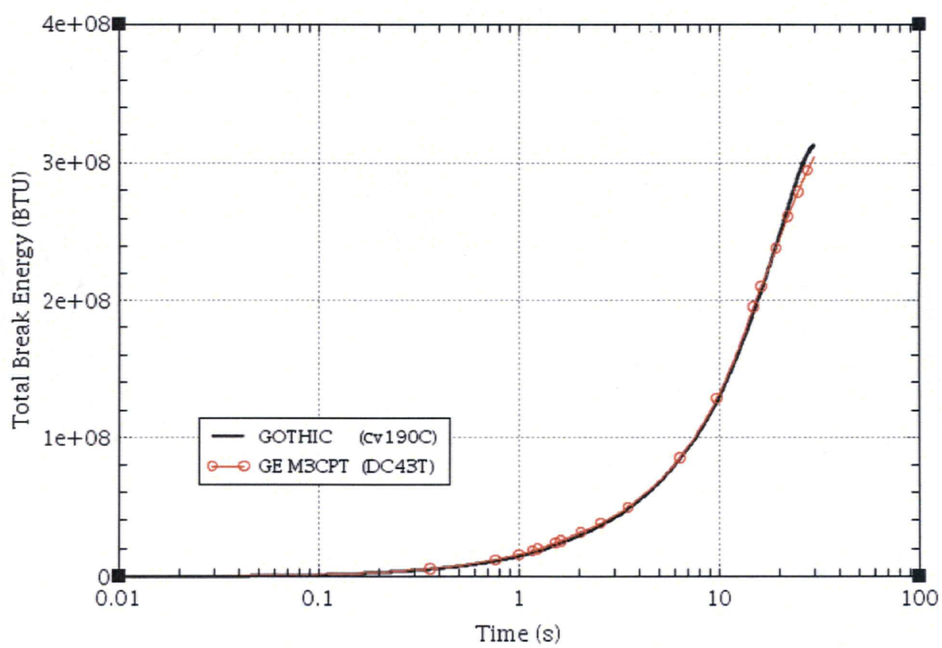


Figure 3: Recirculation Suction Line Break – ST Energy Comparison

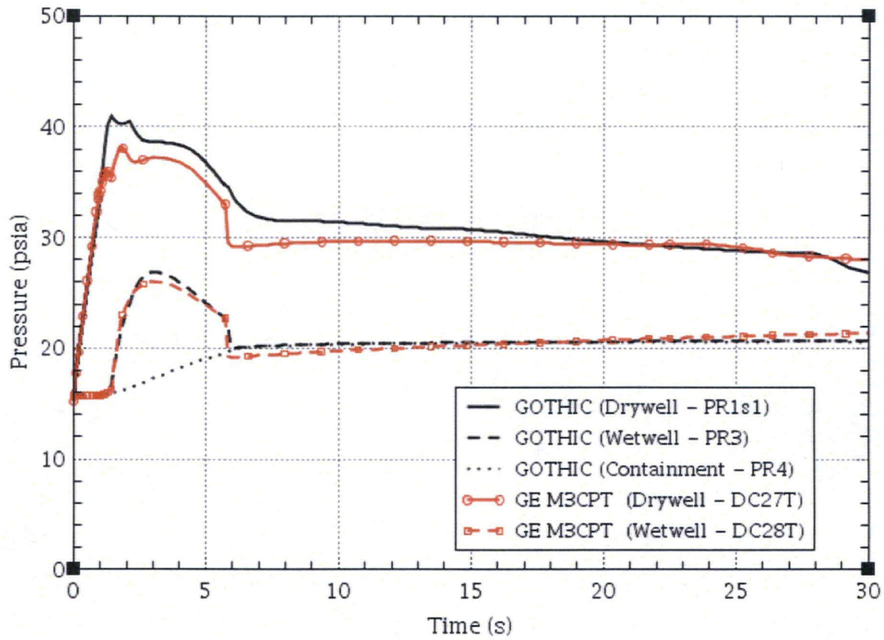


Figure 4: Main Steam Line Break – ST Drywell Pressure Comparison

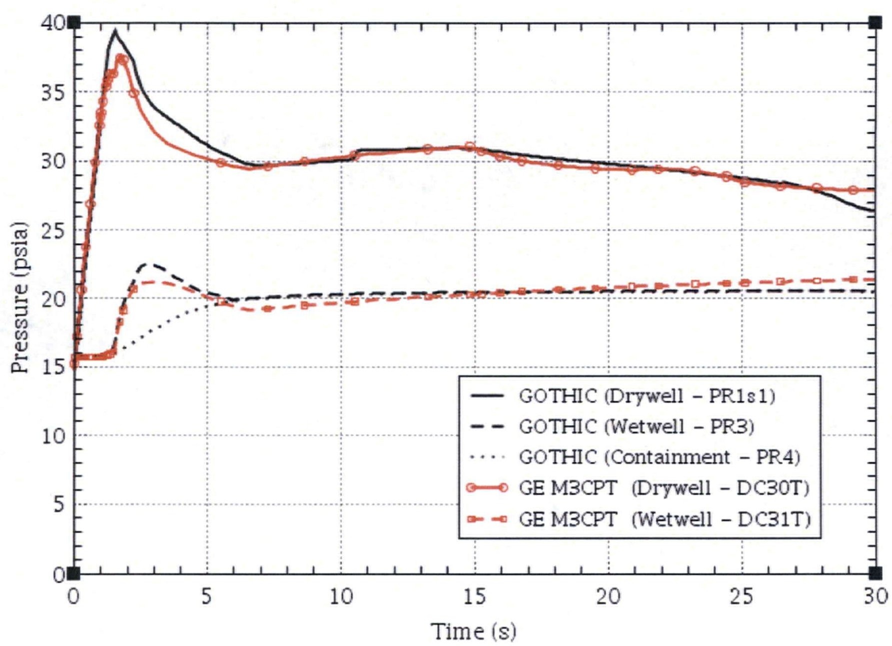


Figure 5: Recirculation Suction Line Break – ST Drywell Pressure Comparison

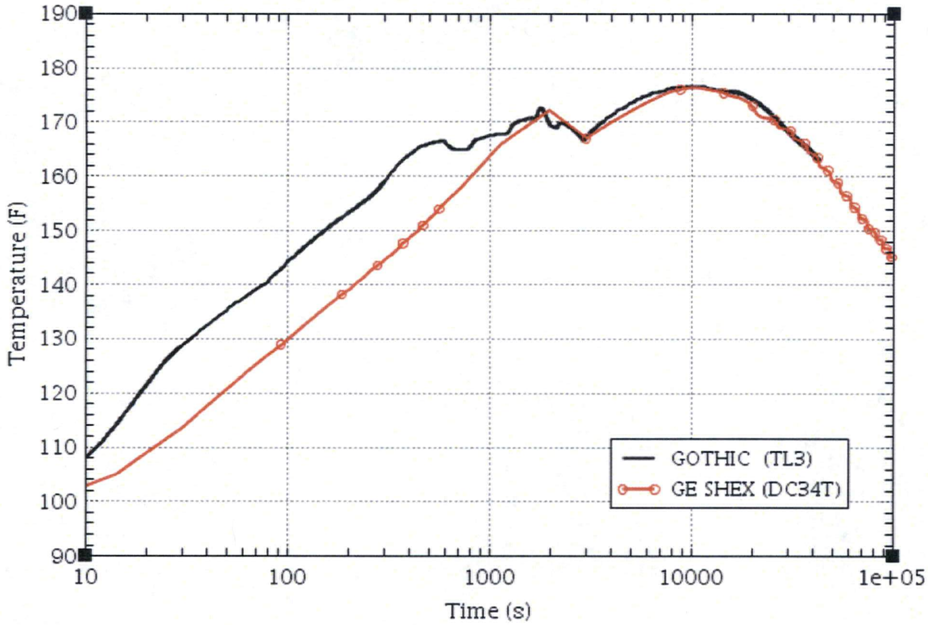


Figure 6: Main Steam Line Break – LT SP Temperature Comparison

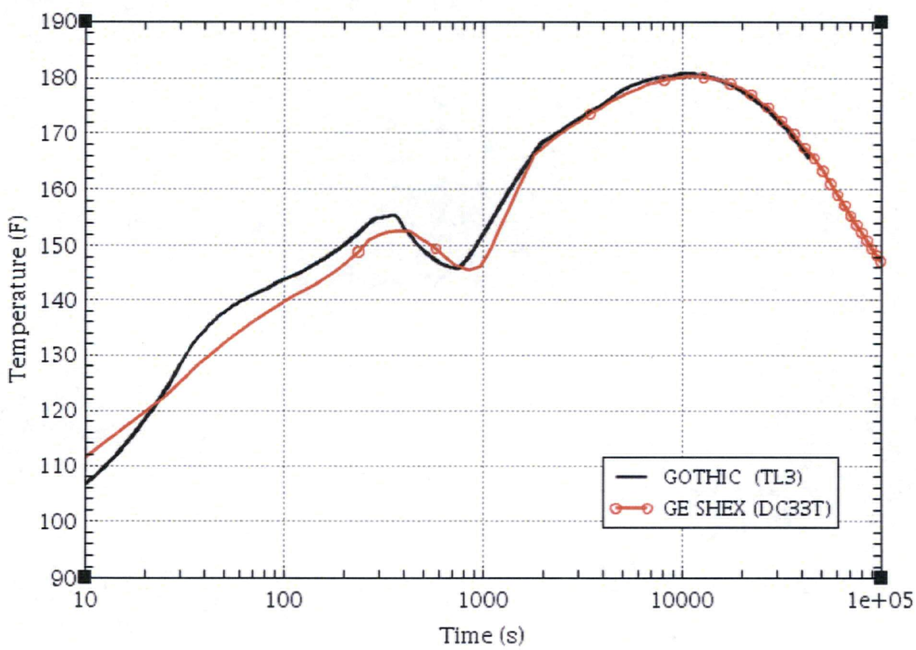


Figure 7: Recirculation Suction Line Break – LT SP Temperature Comparison

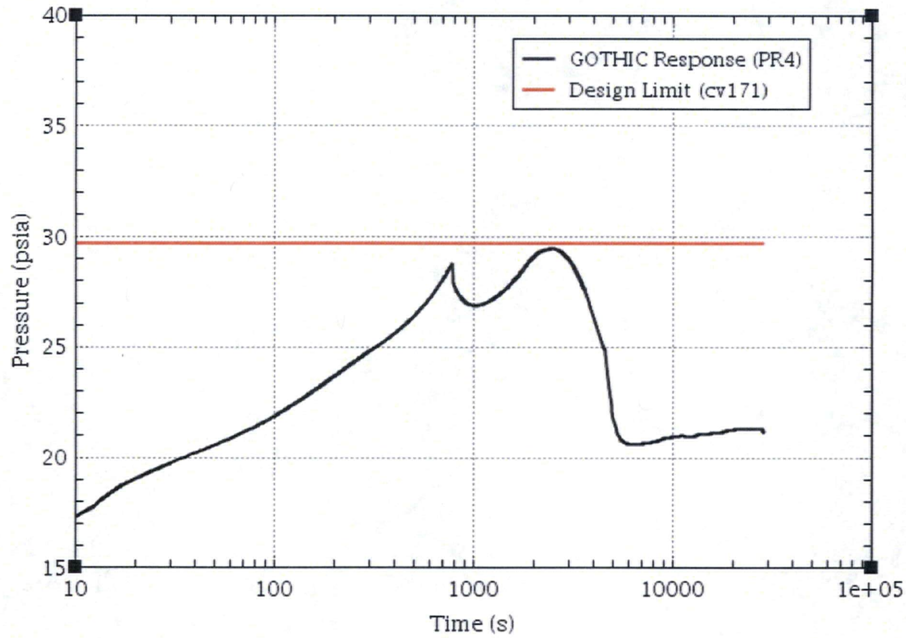


Figure 8: Small Break with Bypass Leakage – LT Containment Pressure Benchmark

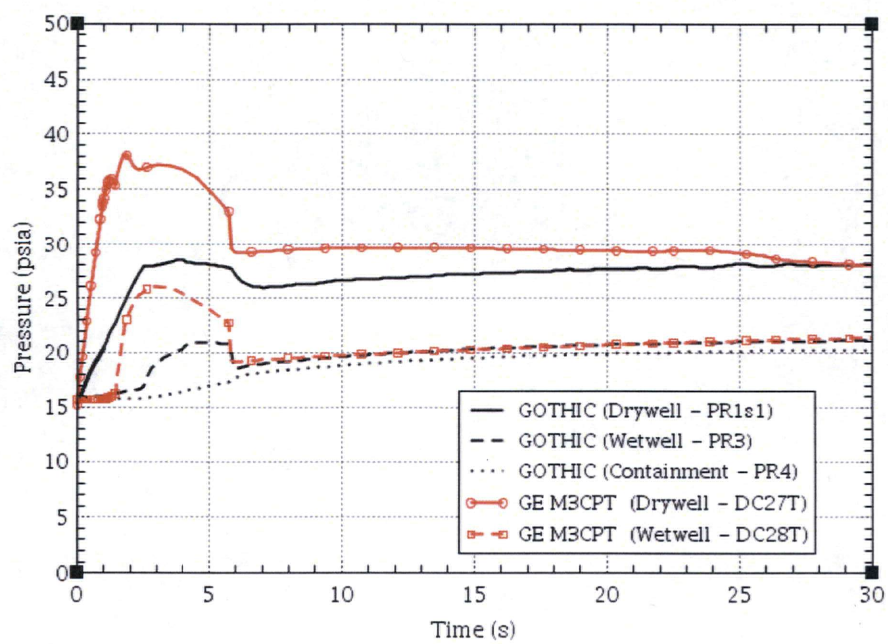


Figure 9: Main Steam Line Break (235 psig RPV) – ST Drywell Pressure

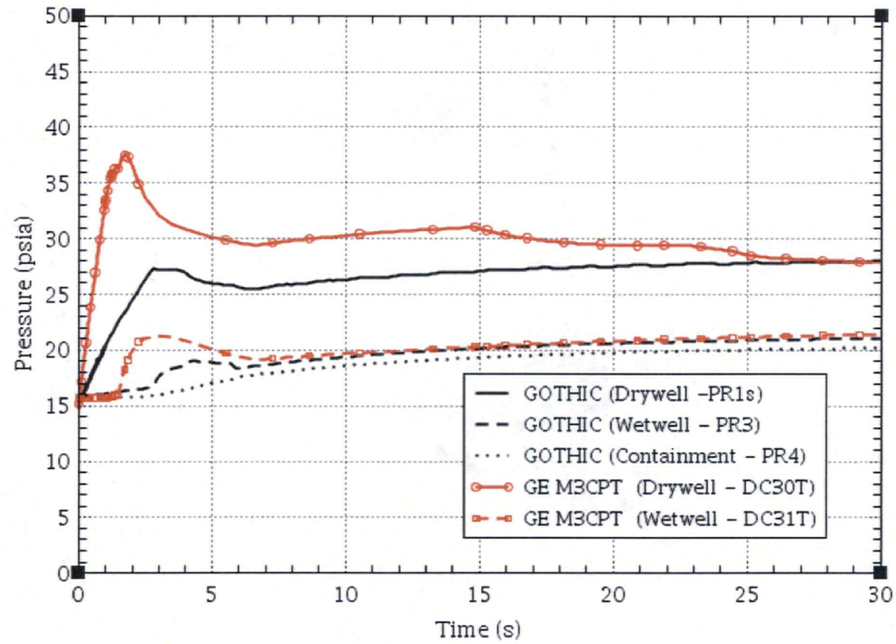


Figure 10: Recirculation Suction Line Break (235 psig RPV) – ST Drywell Pressure

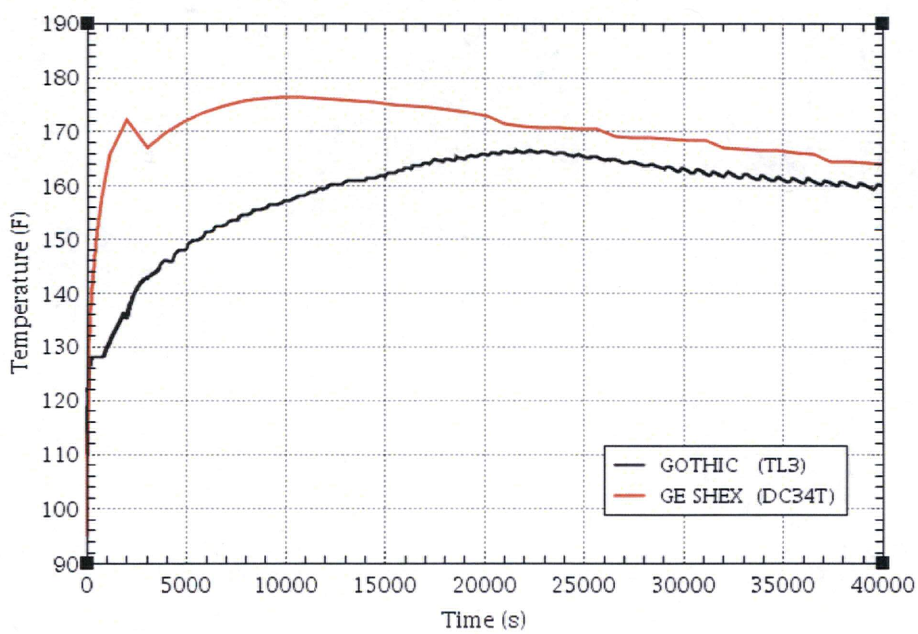


Figure 11: Main Steam Line Break (235 psig RPV) – LT SP Temperature

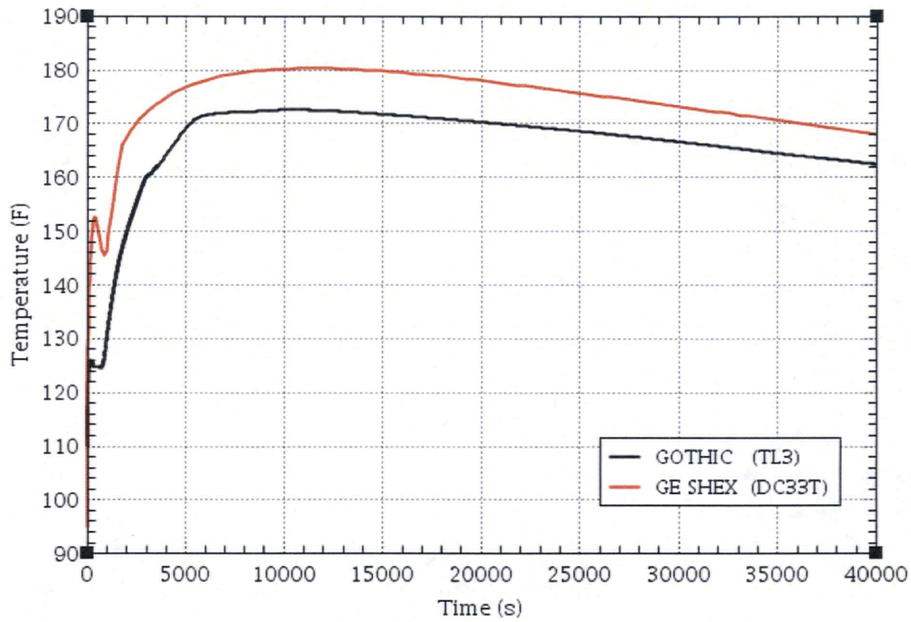


Figure 12: Recirculation Suction Line Break (235 psig RPV) – LT SP Temperature

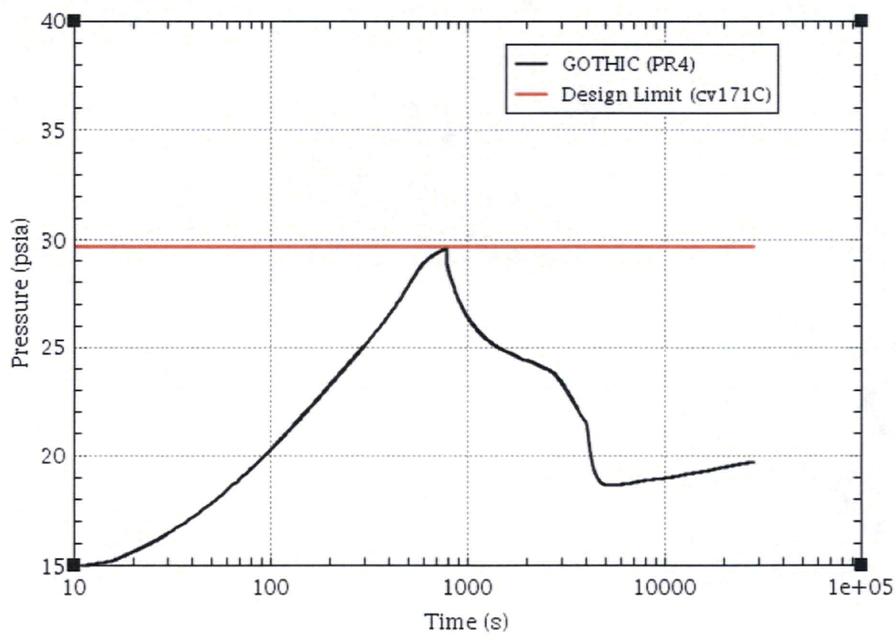


Figure 13: Small Break with Bypass Leakage (235 psig RPV) – LT Containment Pressure

5.0 NOMENCLATURE

BWR	Boiling Water Reactor
CS	Containment Spray
DD	Drain-down
DW	Drywell
EOP	Emergency Operating Procedure
FP	Flow Path
FW	Feedwater
HCU	Hydraulic Control Unit
GOTHIC	Generation of Thermal-Hydraulics Information for Containment Code
HPCS	High Pressure Core Spray
HWL	High Water Level
LOCA	Loss of Coolant Accident
IFTS	Inclined Fuel Transfer System
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LT	Long-Term
LWL	Low Water Level
M3	MODE 3
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
NPSH	Net Positive Suction Head
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RSLB	Recirculation Suction Line Break
SBYP	Steam Bypass
SLLB	Small Liquid Line Break
SPMU	Suppression Pool Makeup System
SSLB	Small Steam Line Break
SP	Suppression Pool
S/RV	Safety / Relief Valve
ST	Short-Term
UCP	Upper Containment Pool
UP	Upper Pool
WW	Wetwell

6.0 REFERENCES

1. PNPP USAR Table 6.2-5, "Initial Conditions Employed in Containment Response Analyses (at 3833 MWt)."

2. PNPP USAR Table 6.2-1, "Key Design and Maximum Accident Parameters for Suppression Containment."

3. PNPP USAR Table 6.2-10, "Available Containment Heat Sinks."

4. PNPP USAR Figure 6.2-25, "Containment Pressure Following a Small Break with Steam Break with Steam Bypass (with Containment Spray and Heat Sinks & a minimum Mark III Design of $A/\sqrt{K}$ = 1.0 ft^2)."