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

February 16, 2017

Mr. David B. Hamilton
Site Vice President
FirstEnergy Nuclear Operating Company
Mail Stop A-PY-A290
P.O. Box 97, 10 Center Road
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT REGARDING TECHNICAL SPECIFICATION CHANGES TO PERMIT INSTALLATION OF UPPER CONTAINMENT POOL GATE AND DRAIN-DOWN OF REACTOR CAVITY PORTION OF UPPER CONTAINMENT POOL (CAC NO. MF7476) (L-16-083)

Dear Mr. Hamilton:

The U.S. Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 174 to Facility Operating License No. NPF-58 for Perry Nuclear Power Plant, Unit No. 1. The amendment consists of changes to the technical specifications (TSs) in response to your application dated March 15, 2016, as supplemented by letters dated November 7, and December 20, 2016, and February 6, 2017.

The amendment revises 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 upper containment pool (UCP) in MODES 1, 2, and 3. The amendment also creates new Special Operations 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.

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

Sincerely,

Kimberly J. Green

Kimberly J. Green, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures:
1. Amendment No. 174 to NPF-58
2. Safety Evaluation

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION, LLC

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. NPF-58

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

   A. The application for amendment filed by FirstEnergy Nuclear Operating Company, et al. (the licensee, FENOC), dated March 15, 2016, as supplemented by letters dated November 7, and December 20, 2016, and February 6, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

    (2) <u>Technical Specifications</u>

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 174, are hereby incorporated into this license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-58 and
Technical Specifications

Date of Issuance: February 16, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 174

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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

REMOVE	INSERT
License NPF-58	License NPF-58
- 4 -	- 4 -
TSs	TSs
iv	iv
3.6-39	3.6-39
3.6-43	3.6-43
---	3.6-44
---	3.10-23
---	3.10-24
---	3.10-25
---	3.10-26

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at reactor core power levels not in excess of 3758 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 174, are hereby incorporated into the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

a. FirstEnergy Nuclear Generation, LLC

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.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.2.1	Verify suppression pool water level is within limits.	In accordance with the Surveillance Frequency Control Program

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.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.4.2	Verify upper containment pool water temperature is ≤ 110° F.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.4.3	Verify each SPMU subsystem manual power, power operated, and automatic valve that is not locked, sealed, or otherwise secured in positon is in the correct position.	In accordance with the Surveillance Frequency Control Program

(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.	In accordance with the Surveillance Frequency Control Program
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.	In accordance with the Surveillance Frequency Control Program

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

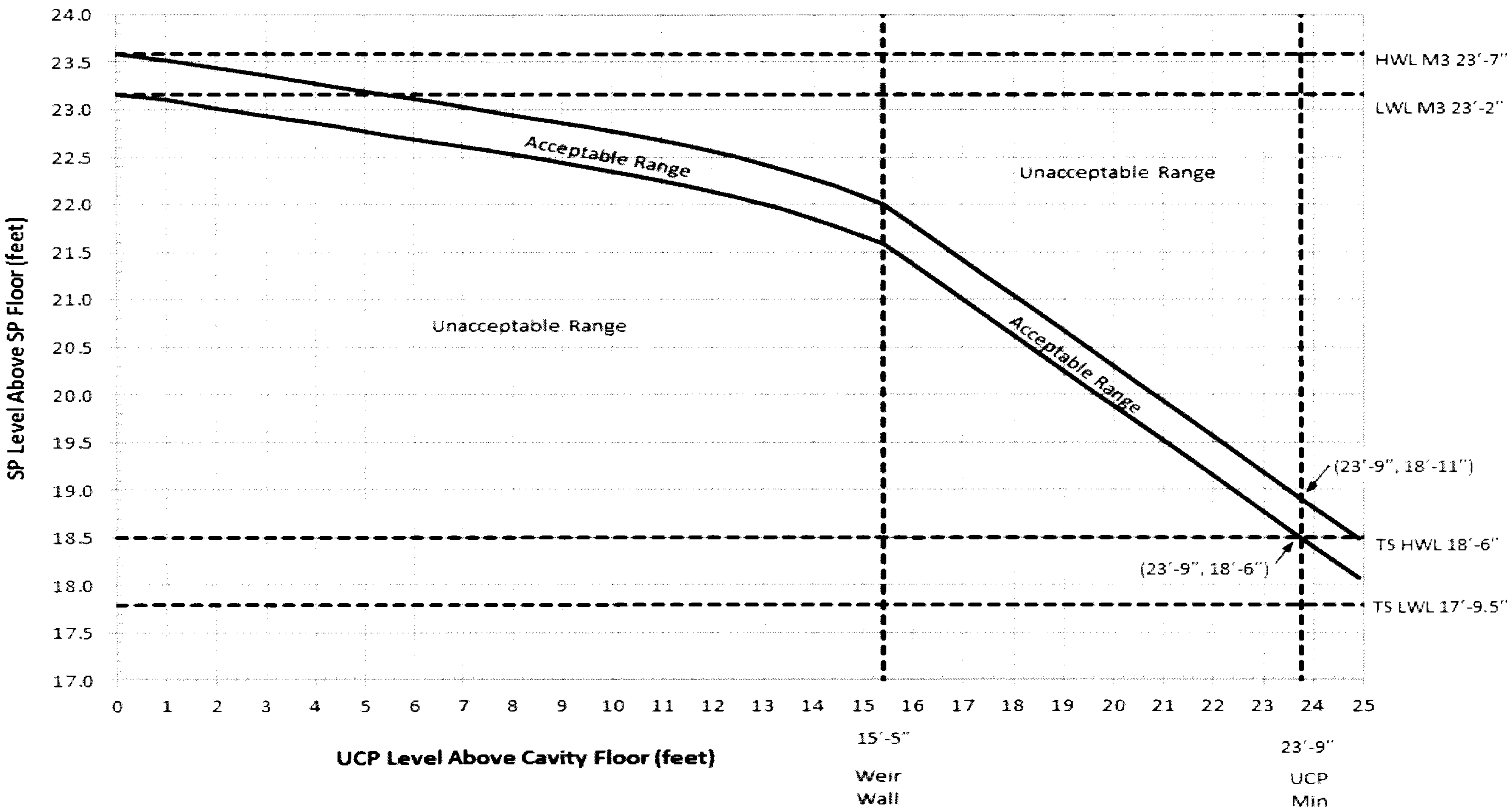
---NOTE---
Separate Condition entry is allowed for each requirement of the LCO.

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.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION, LLC

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By application dated March 15, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16075A411), as supplemented by letters dated November 7, and December 20, 2016, and February 6, 2017 (ADAMS Accession Nos. ML16312A359, ML16356A005, and ML17038A114, respectively), FirstEnergy Nuclear Operating Company (the licensee or FENOC) requested changes to the technical specifications (TSs) for the Perry Nuclear Power Plant, Unit 1 (PNPP or Perry). The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 10, 2016 (81 FR 28898).

The proposed changes would revise 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 upper containment pool (UCP) in MODES 1, 2, and 3. The proposed amendment would also create new Special Operations 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.

2.0 REGULATORY EVALUATION

2.1 Description of the Upper Containment Pool and its Safety Function

Perry is a boiling-water reactor (BWR)/6 with a Mark III containment. The containment building encloses the drywell which is a cylindrical, reinforced concrete structure with a removable head. The reactor cavity portion of the UCP lies above the drywell head. The drywell encloses the reactor vessel and the reactor coolant system (RCS). It is designed to withstand the pressure and temperature of the steam generated by a RCS pipe rupture, and channel the steam to the suppression pool via horizontal vents located in the drywell wall. The suppression pool contains a large volume of water which rapidly condenses steam directed to it. Initially, the air in the

drywell is forced into the suppression pool by the steam discharging from the postulated break and pressurizes the containment.

The drywell provides the structural support for the UCP. The UCP provides several functions: (1) radiation shielding when the reactor is in operation, (2) storage space for the dryer, separator, and fuel assemblies during refueling, (3) an area for fuel transfer during refueling, (4) storage for control blade guides, new blades, refueling tools, and other irradiated and unirradiated components (fuel assemblies are not stored in the UCP during operation), and (5) the UCP provides water to the suppression pool following a loss-of-coolant accident (LOCA) by means of the 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 water level signal or a timer set for 30 minutes after the LOCA begins.

Perry's 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 safety relief valve (SRV) quenchers, the horizontal drywell vents, and the reactor core isolation cooling (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 keeps the suppression pool at an acceptable water level. In order to ensure the proper amount of water is transferred 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 for all safety-related functions of the suppression pool. These functions include: (1) providing ECCS with a source of water for injection into the vessel following a LOCA, (2) providing a heat sink for the decay and sensible heat released during reactor blowdown from the SRVs or from a LOCA, (3) providing adequate NPSH to the ECCS pumps, (4) condensing steam discharged from the RCIC system turbine, (5) providing a long term heat sink for cooldown of the reactor, and (6) maintaining structural loads on the drywell and containment structures within acceptable limits.

The minimum suppression pool water level limit ensures adequate coverage of the horizontal vents during the initial portion of the LOCA. This ensures that steam discharged from the SRV quenchers, main vents, and the RCIC turbine exhaust lines is completely condensed. The ECCS takes suction from the suppression pool and the water is injected into the reactor vessel and spills out of the break. This water forms a pool in the bottom of the drywell inside the weir wall. This pool is referred to as the drywell pool. The water in the drywell pool is not available to the suppression pool until the water level rises sufficiently to overflow the drywell weir wall; this overflow returns to the suppression pool. The water inside and below the top of the weir

wall remains unavailable, as well as water entrapped in other volumes. This entrapped water reduces the volume of water in the suppression pool; this is referred to as suppression pool drawdown. The sizing of the UCP provides sufficient water to the suppression pool to maintain the minimum pool level two feet above the top row of vents, considering the entrapped water which cannot return to the suppression pool.

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

The maximum suppression pool water level limit ensures that clearing loads from the SRV 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.

2.2 Proposed Changes to the Perry Technical Specifications

The licensee has proposed the following changes to the TSs regarding the water level in the suppression pool and upper containment pool.

TS 3.6.2.2: The licensee proposes the following revision to Limiting Condition for Operation (LCO) 3.6.2.2 to add a new suppression pool water level range when the reactor well to steam dryer storage pool gate is installed in MODES 1, 2, and 3:

> , 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.

TS SR 3.6.2.4.1: The licensee proposes the following addition to Items a and b, and the addition of Item c to the upper containment pool water level surveillance requirement to maintain the upper pool water level and the suppression pool water level when the reactor well to steam dryer storage pool gate is installed:

a. , when the reactor well to steam dryer storage pool gate is not installed.

b. , 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.

TS SR 3.6.2.4.4: The licensee proposes to add the following note to the SR:

> 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 [inclined fuel transfer system] carriage is located in the upper pool, and IFTS transfer tube shutoff valve 1F42F002 is closed.

TS 3.10.9: The licensees proposes the creation of new TS 3.10.9, "Suppression Pool Makeup-MODE 3 Upper Containment Pool Drain-Down," including Figure 3.10.9-1, 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.

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		FREQUENCY
SR 3.10.9.1	Verify suppression pool temperatures 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," is also added, but is not reproduced here. Figure 3.10.9-1 provides the acceptable range of suppression pool level above the suppression pool floor (in feet) as a function of UCP level above the cavity floor (in feet).

2.3 <u>Proposed Changes to the Design Basis</u>

The licensee has proposed changes to 3 of the 12 design basis criteria used for the SPMU as currently described in updated safety analysis report (USAR) Section 6.2.7.1. The design basis criteria are discussed in the license amendment request (LAR) Section 3.1.5. The three proposed changes are described below.

2.3.1 Minimum Normal Operation Low Water Level (LWL)

The suppression pool LWL will need to be increased by 5.4 inches + 0.3 inches (accounting for level instrument uncertainty) to a new lower limit of 18 feet 3.2 inches when the reactor well to steam dryer storage pool gate is installed in MODES 1, 2, and 3. If the MODE 3 drain-down of the reactor cavity portion of the UCP is conducted, the suppression pool LWL will need to be increased in accordance with new TS Figure 3.10.9-1.

2.3.2 Maximum Normal Operation High Water Level (HWL)

If the MODE 3 drain-down of the reactor cavity portion of the UCP is conducted, the suppression pool HWL will need to be increased in accordance with new TS Figure 3.10.9-1.

2.3.3 Post-Accident Entrapment Volumes

As described in LAR Sections 3.1.5, 3.2.1, and 3.2.2, adjustments to the existing entrapment volumes are needed when the reactor well gate is installed in MODES 1, 2, and 3, and when the reactor cavity portion of the UCP is drained in MODE 3.

2.4 Applicable Regulatory Requirements and Guidance

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

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

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

GDC 50, "Containment Design Basis," requires that the reactor containment structure and its internal compartments accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. The margin must include conservatism of the calculation model and input parameters.

Section 182a of the Atomic Energy Act, as amended (the "Act"), requires applicants for nuclear power plant operating licenses to incorporate TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are set forth in 10 CFR 50.36. That regulation requires that the TSs include items in five categories, including: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation, (3) SRs, (4) design features, and (5) administrative controls. It also states that the Commission may include such additional TSs as it finds to be appropriate. However, the regulation does not specify the particular TS to be included in a plant's license.

The regulations in 10 CFR 50.36(c)(2) require that TSs include LCOs. LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the

facility. When LCOs are not met, the licensee shall shutdown the reactor or follow the remedial action permitted by the technical specifications. The regulations at 10 CFR 50.36(c)(2)(ii) state that LCOs must be established for each item meeting one of four criteria:

> *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> *Criterion 4.* A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The regulation at 10 CFR 50.36(c)(3) requires TSs to include items in the category of SRs, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

The NRC staff's guidance for review of TSs is in Chapter 16, *Technical Specifications*, of NUREG-0800, Revision 3, *Standard Review Plan* (March 2010) (ADAMS Accession No. ML100351425).

Title 10 of the *Code of Federal Regulations* Part 50.34, "Contents of application; technical information", Section (b)(6) provides the information concerning facility operation that is required in a final safety analysis report.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 13.5.2.1, Revision 2, "Operating and Emergency Operating Procedures," describes the operating procedures that will be used by the operating organization to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner (ADAMS Accession No. ML070100635).

NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, provides the basis for performing a human factor review for license amendments (ADAMS Accession No. ML12324A013).

3.0 TECHNICAL EVALUATION

3.1 Use of the GOTHIC Code

The licensee's LAR is based, in part, on analyses performed using the GOTHIC computer code. The Electric Power Research Institute (EPRI) developed and maintains GOTHIC for

use in containment and other multi-dimensional two-phase flow applications. 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.

GOTHIC has not been used as part of the PNPP licensing basis prior to this proposed TS change. However, the GOTHIC code has been used in the past by the nuclear power industry, and the NRC has previously accepted analyses that used GOTHIC.

In order to evaluate the conservatism used in the PNPP GOTHIC calculation, in the request for additional information (RAI) 1 (ADAMS Accession No. ML16279A043), the NRC staff asked the licensee to describe the containment spray system (CS) actuation timing and summary of inputs used in calculating CS set point, and to confirm that the GOTHIC analyses performed in support of the LAR do not impact the CS set points described in Section 6 of the PNPP USAR. In response (ADAMS Accession No. ML16312A359), the licensee indicated the PNPP CS setpoint methodology utilizes, NEDC-31336, "General Electric [GE] Instrument Setpoint Methodology," (ADAMS Accession No. ML073450560) that is approved by the NRC and described in the PNPP USAR Section 6.2.1.1.5.4.

In RAI 2 (ADAMS Accession No. ML16279A043), the NRC staff asked the licensee to describe the conservatisms used in the input assumptions and the benchmarking performed for the GOTHIC model for the steam line break with steam bypass of the suppression pool transient, as referenced in the LAR. In response (ADAMS Accession No. ML16312A359), the licensee listed several assumptions, including:

- The GOTHIC model uses the same heat sinks as used in the GE model (surface areas and volumes as shown in USAR, Table 6.2-10).
- The suppression pool level is at the technical specification maximum level. The additional static head results in a higher differential pressure between the drywell and containment, causing more steam to leave the drywell through the leakage path, which results in a higher containment pressure.
- Break steam flow is computed by GOTHIC using a TABLES setting for the critical flow model. Since the break flow is predominantly dry saturated steam, the TABLES setting in GOTHIC will use a combination of the Moody model and an isentropic ideal gas model to compute the critical flow limit for break flow. The steam bypass flow path also uses a TABLES setting for the critical flow model. A plot of the steam bypass flows confirms no liquid or drop flow through the bypass flow path. Both paths conservatively inhibit the cooling effect of drops in the drywell and in containment.
- The GOTHIC model includes heat from the operating containment spray pump, which results in a conservatively higher temperature spray.
- The GOTHIC model includes modeling the RPV with decay heat from the fuel and metal heat sinks. This results in conservatively maximizing the heat, and, thus, the steam source of the RPV.

The NRC staff's audit of the licensee's GOTHIC analyses in support of this LAR (ADAMS Accession No. ML16309A081), confirmed the validity of the above information, and confirmed

that the licensee's benchmarking analyses performed in support of the MODE 3 UCP drain down (see SE Section 3.5): (1) have incorporated input assumptions similar to the ones used in the PNPP's USAR in a manner to predict conservative containment temperature and pressure, and (2) produced results very close to the licensee's design-basis accident (DBA) described in PNPP's USAR.

In addition, the GOTHIC code has been successfully validated against a wide variety of data and analytical problems. The GOTHIC code has been benchmarked in the past by the industry against test data and found to provide conservative results (ADAMS Accession No. ML062850149).

Based on the above information, the NRC staff concludes that the PNPP's GOTHIC input assumptions and modeling provided in support of this LAR, produced conservative results for factors such as containment temperature, pressure and suppression pool level, and, therefore, is acceptable for use for this application.

3.2 Hydrodynamic Load Consideration

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 SRV actuations.

As detailed in Section 3.2.2 of the LAR, the licensee determined that the total design basis entrapment volume is increased for drain-down conditions in MODE 3. To offset the increase in entrapment volume, volumes from the UCP steam separator storage pool and a drawdown volume above the 2-foot vent submergence post-accident requirement (for the current TS LWL limit) are available. The total inventory that needs to be added to the suppression pool under MODE 3 conditions to provide 2 feet of vent coverage corresponds to a lower bound analytical limit of 23 feet 1 inch. Adding 1 inch to account for suppression pool level instrument uncertainty, and adding an operating range of 5 inches, the new suppression pool level range is ≥ 23 feet 2 inches and ≤ 23 feet 7 inches.

Because the licensee will maintain the SPMU design basis criterion of 2-foot post-accident vent submergence, the NRC staff finds that the change to the entrapment volume and to the suppression pool level range acceptable.

The licensee performed analyses on the hydrodynamic loads in the containment due to a primary system pipe break. The evaluations considered the impact of a maximum suppression pool level of at least 23 feet 8 inches, which is 5 feet 2 inches above the current TS HWL limit of 18 feet 6 inches, and includes 1 inch for measurement uncertainty above the new suppression pool level range.

The NRC staff audited the licensee's GOTHIC analysis related to the containment loads generated during the first part of the limiting LOCA in MODE 3, which is a main steam line break (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, approximately at 5 feet 2 inches above the current TS HWL limit (ADAMS Accession No. ML16309A081). Performing the calculations at the higher level is conservative because the larger volume produces the greater impacts on the hydrodynamic loads. The staff's review during the audit confirmed a conservatively calculated peak drywell pressure of 13.8 psig [pound per square inch gauge] (28.53 psia [pound per square inch absolute]) for the MODE 3 limiting LOCA. 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.

As discussed in Section 3.1 of this SE, the NRC staff has determined that the GOTHIC analysis performed in support of this LAR is conservative. Because the containment loads for the postulated MODE 3 drain-down conditions are bounded by the design basis accident for MODE 1, which is evaluated at full power operating pressure, the staff concludes that the plant will continue to meet GDC 4 and GDC 50.

3.2.1 Water Jet Loads

In LAR Section 3.3.1, the licensee states that 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.

All of these loads are primarily a function of the drywell pressure. For the low pressure LOCA in MODE 3, the NRC staff's audit of the licensee's analysis (ADAMS Accession No. ML16309A081) confirmed that the loads are smaller than the DBA loads due to the reduced drywell pressure and that the increased liquid level in the suppression pool has no additional impact on these loads. The containment pressure will continue to meet GDC 50, and, therefore, the calculated water jet loads are acceptable.

3.2.2 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 (PNPP USAR Chapter 15, ADAMS Accession No. ML15314A182, not publically available), 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 for the lower pressure events and, therefore, produce lower pressure differentials.

3.2.3 Pool Swell Drag and Impact Loads

In Amendment No. 100 for PNPP, the NRC staff concluded that the impact loads are a function of the pool surface velocity and the pool swell rate is a function of the air flow through the vents (ADAMS Accession No. ML021840212). Therefore, for the low pressure LOCA, the drywell pressure feeding the bubble is smaller than the DBA drywell pressure and the resultant vent flow rates and the ultimate bubble size in a low pressure LOCA will be smaller than that experienced during a DBA. That results in smaller bubble growth rate and lower impact and drag loads. In addition, pool swell velocity is substantially reduced if there is venting through only one row of vents as opposed to two rows of vents.

The NRC staff's audit of the licensee's GOTHIC analysis (ADAMS Accession No. ML16309A081) at low pressure event in MODE 3, confirmed that most of the venting occurs through the top row of vents with only a short period of flow passing through the second row of vents and none through the bottom row of vents

Therefore, the NRC staff concludes 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 the full-power MODE 1 DBA analysis results. The containment pressure will continue to meet GDC 50, and, therefore, is acceptable.

3.2.4 Fallback Loads

In the LAR, the licensee states that the fallback loads are created due to the physical phenomena that, after the bubble breaks through the pool surface, water will "fall back" to the pool imposing impact and drag loads on equipment and structures.

As explained above, the LOCA in MODE 3 experiences lower pressure conditions compared to LOCA in MODE 1 (DBA LOCA) therefore, the pool swell is smaller than that experienced for the DBA LOCA (MODE 1). The lower pool swell at lower pressure conditions will results in smaller "maximum velocity of the falling water impact loads" and the drag loads in the lower pressure MODE 3 events. This is consistent with the licensee's position regarding the fallback loads.

In addition, the NRC staff's audit of GOTHIC analysis of LOCA in MODE 3 (ADAMS Accession No. ML16309A081) and the review of PNPP USAR Chapter 15 supports that the lower pool swell at lower pressure conditions will result in smaller "maximum velocity of the falling water impact loads" and the drag loads in the lower pressure MODE 3 events, and, therefore, is acceptable.

3.2.5 Froth Impingement and Drag Loads

In the LAR, the licensee states that, 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 floor. The NRC staff recognizes that 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 exists less froth and the maximum froth level is lower during the low pressure events. Therefore, the NRC staff concludes that the loads are all bounded by the DBA.

3.2.6 Condensation Oscillation Loads

In the LAR, the licensee states that once the vents have cleared and the pool swell transient has passed, there could be a period where the surface of the steam bubble that forms in the pool just beyond the top vent oscillates at low frequencies causing cyclic loading on submerged pool structures and boundaries.

The condensation oscillation (CO) loads can occur in the low pressure LOCA as well as the DBA. 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. GE developed load and frequency functions that consider these effects and are bounding for all break events. In addition, the strength of the pressure pulses at the upper vent is independent of the vent submergence and vent submergence is not a parameter in the GE methodology accepted by

the NRC for calculating CO loads on equipment (PNPP USAR Appendix 3B, "Containment Loads," ADAMS Accession No. ML15314A168, not publically available). Therefore, the NRC staff concludes that the DBA CO load function is bounding for the low pressure LOCA.

3.2.7 Chugging Loads

In the LAR, the licensee states that, when the steam flow through the top vents falls below 10 lbm/ft^2-s (pound mass/square feet-second), the oscillatory condensation turns to an erratic chugging mode with a pressure pulse generated. The NRC has previously determined that load analysis for chugging for the DBA covers all break sizes (ADAMS Accession No. ML021840212) and that chugging in the low pressure LOCA is not expected to be significantly different from other cases previously considered. In addition, 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 (PNPP USAR Appendix 3B, "Containment Loads").

Supplement 8 to the Safety Evaluation Report (SER) related to the operation of Perry Nuclear Power Plant, Units 1 and 2, dated January 1986, included Humphrey Concern 19.1 (ADAMS Accession No. ML091320307), which identified that increased vent submergence could cause an increase in chugging loads. A technical evaluation was performed. The transmittal 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 (hertz) 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 Supplement 8 to the Perry SER. The NRC SE disposition, which documents Humphrey Concern 19.1, envelopes the chugging loads for the proposed increase in the suppression pool level for MODE 3 drain-down.

3.2.8 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 the containment temperature is at the suppression pool temperature to maximize the containment pressure (PNPP USAR Appendix 3B, "Containment Loads").

The NRC staff's audit of the PNPP GOTHIC analysis (ADAMS Accession No. ML16309A081) confirmed that during the drain down in MODE 3, the energy deposited to the suppression pool is less than in the DBA (MODE 1), which resulted in lower containment pressure and, therefore, lower depressurization loads in MODE 3. As discussed in Section 3.1 of this SE the staff determined that the PNPP GOTHIC modeling and input assumptions supported by the benchmarking used for this LAR results in a conservative containment temperature and pressure that are the primary factors for the calculated drywell depressurization loads.

Therefore, the licensee's GOTHIC analysis results related to drywell depressurizations loads are acceptable and are in compliance with GDC 4 as discussed in Section 2.4 of this SE.

3.2.9 Safety Relief Valve Actuation Loads

Hydrodynamic loads from the SRV 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 5 feet over normal suppression pool high water level, on SRV loads was previously addressed with the resolution of Brookhaven National Laboratory (BNL) concerns BNL-2 and BNL-3 as documented in Supplement 8 to the Perry SER. The NRC staff's review of the SER indicates that the SRV discharge line thrust loads are within the capability of the existing SRV discharge line configurations. Therefore, the NRC staff concludes that during MODE 3 drain-down conditions, the loads from an SRV lift will be less than the design values.

Based on the above evaluations in Sections 3.2.1 through 3.2.9, the NRC staff concludes that the containment loads at the MODE 3 drain-down conditions are bounded by the design basis analyses evaluated at full power operating pressure, and are in compliance with GDC 50.

3.3 Net Positive Suction Head of the Emergency Core Cooling System Pumps

As discussed in LAR Section 3.4, the licensee analyzed the effects of the proposed changes to the suppression pool and UCP levels on the NPSH. The ECCS pumps, including the low pressure coolant injection, high pressure core spray, and low pressure core spray system pumps, have been analyzed for NPSH requirements in the Perry USAR Chapters 5 and 6 (ADAMS Accession Nos. ML15314A171 and ML15314A172, not publically available). The analyses are performed assuming 212 °F [degree Fahrenheit] 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, and is sufficient to eliminate concerns such as vortexing, flashing, and cavitation during a LOCA which are dependent on suppression pool level analysis results.

As discussed in Section 3.1 of this SE, the NRC staff determined that the PNPP GOTHIC modeling and input assumptions supported by the benchmarking used for this LAR results in a conservative containment temperature and pressure. This supports that the GOTHIC analysis results in a conservative suppression pool level and will not adversely impact the NPSH calculation results. Based on the discussions provided in Section 3.1 of this SE, the NRC staff finds that the GOTHIC analysis results in a conservative pool level.

Therefore, the licensee's GOTHIC analysis results related to drywell depressurizations loads are in compliance with GDC 4 and GDC 16, and, therefore, are acceptable.

3.4 Long Term Heat Sink

The suppression pool volume provides a long term heat sink for the decay and sensible heat released during a LOCA. In the LAR, the licensee states that 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.

As detailed in Section 3.2.1 of the LAR, the licensee determined that reductions in the entrapment volumes are needed when the reactor well to steam dryer storage pool gate is installed during MODES 1, 2 and 3. The decrease in the SPMU inventory is not fully offset by

the reduction in the total design basis entrapment volume. Therefore, to ensure that the SPMU design basis criterion of maintaining a 2-foot post-accident vent submergence, the suppression pool TS LWL must be increased by 5.4 inches to a new analytical lower limit of 18 feet 2.9 inches. This amount is further increase by 0.3 inches to 18 feet 3.2 inches to account for level instrument uncertainty. The suppression pool TS HWL remains unchanged for gate installation in MODES 1, 2, and 3.

As stated by the licensee, the combined water inventory between the UCP and the suppression pool will 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.

Because the licensee will maintain the SPMU design basis criterion of 2-foot post-accident vent submergence, the NRC staff finds that the changes to the entrapment volumes and the increase in the suppression pool TS LWL are acceptable.

The NRC staff audited the GOTHIC simulations performed by the licensee considering the MSLB and recirculation suction line break DBAs with the reactor well drained (ADAMS Accession No. ML16309A081). The initial suppression pool water level in the GOTHIC simulations for the long-term events is at the current TS LWL. Performing the calculation at the lower (current) suppression pool LWL is conservative because a higher LWL means that more inventory is available for long term heat removal. The staff's audit confirmed the licensee's GOTHIC analysis results indicate a peak long term suppression pool temperature with about a 12 °F margin between the calculated pool temperature and the design temperature limit of 185 °F exists.

During the audit (ADAMS Accession No. ML16327A018), the NRC staff questioned the conservatisms used in the GOTHIC analysis performed by the licensee in support of the LAR related to steam line break. In response, the licensee indicated the GOTHIC model incorporated input assumptions similar to the PNPP's licensing-basis methodology, which complies with the GDC 50 requirements, and has been previously approved by the NRC staff (ADAMS Accession No. ML16312A359). Other conservatisms used in the GOTHIC analysis are discussed in Section 3.1 of this SE.

In summary, the NRC staff's review of the GOTHIC analysis confirmed that 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 (i.e., 185 °F).

Therefore, the NRC staff concludes that the licensee's GOTHIC analysis results for long term suppression pool temperature is below the design temperature limit, and, as such, will continue to meet GDC 38. Therefore, the staff finds this analysis to be acceptable.

3.5 Steam Line Break with Steam Bypass of Suppression Pool

As described in LAR Section 3.7, the design of the pressure suppression reactor containment is such 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 automatically initiates the containment spray system any time after LOCA plus 10 minutes (PNPP USAR Chapter 15, ADAMS Accession No. ML15314A182, not publically available).

The licensee analyzed the impact of raising the suppression pool level on the steam bypass capability at a reduced vessel pressure of 235 psig by performing a GOTHIC analysis. As discussed in the LAR, the licensee performed a benchmark analysis of the GOTHIC analysis against the bypass capability analysis described in USAR Section 6.2.1.1.5.4, using the same design basis conditions: containment spray is activated 180 seconds after containment pressure reaches 9 psig or at LOCA plus 13 minutes, whichever occurs later (PNPP USAR Chapter 15). This yielded the limiting break size (0.5 ft^2) that would produce the limiting containment pressure response that satisfied the containment design pressure limit of 15 psig. The licensee then performed the GOTHIC analysis for a containment pressure response during a steam bypass event for MODE 3 drain-down conditions with a vessel pressure of 235 psig and the design basis conditions listed above. For conservatism, the initial suppression pool level was set in excess of the upper bound analytical limit. For the limiting break size (0.5 ft^2), the results showed that the expected peak containment pressure is 29.66 psia (14.96 psig), which is below the design limit of 29.70 psia (15 psig).

In conjunction with GDC 16 requirements to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions for systems important to safety are not exceeded for as long as postulated accident conditions require, the NRC staff asked the licensee to explain the CS actuation timing and the input assumptions used in the GOTHIC calculations in support of this LAR (ADAMS Accession No. ML16279A043).

In response, the licensee indicated the PNPP CS setpoint methodology utilizes NEDC-31336, "General Electric Instrument Setpoint Methodology," which was approved by the NRC, as reflected in the PNPP USAR Section 6.2.1.1.5.4 (ADAMS Accession No. ML16312A359, not publicly available). Because the CS actuation set point and timing are not being changed by this amendment request, the staff finds that the MODE 3 drain-down configuration will meet GDC 16, and, therefore, is acceptable.

The NRC staff audited the GOTHIC analysis for the containment pressure response during a steam bypass event in a MODE 3 drain-down configuration (ADAMS Accession No. ML16309A081). The staff reviewed the input assumptions and calculation, and confirmed that the licensee conservatively calculated the peak containment pressure for the most limiting break size.

As discussed in Section 3.1 of this SE, the NRC staff has determined that the GOTHIC analysis performed in support of this LAR, specifically the containment temperature and pressure calculation, is conservative. The steam line break with steam bypass of suppression pool pressure analysis results in a pressure lower than the containment design pressure. The NRC staff concludes that the plant design will continue to meet GDC 50.

3.6 Upper Containment Pool Dump Time vs. 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. The NRC staff audited the licensee's analysis of the SPMU dump time for operations with the gate installed and with the reactor well pool drained (ADAMS Accession No. ML16309A081). 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 less than the pump time. The staff reviewed the analysis results, and confirmed for the reactor well pool drained in MODE 3, the allowable dump time is less than the pump time. Therefore, the NRC staff concludes the SPMU "dump time" meets GDCs 16 and 50, and, therefore, is acceptable.

3.7 <u>Human Factors Consideration</u>

3.7.1 Operator Actions

Installing the reactor well gate in MODES 1, 2, and 3 will isolate the water in the Steam Dryer Storage Pool from the SPMU system. In the Enclosure to letter dated March 15, 2016, Section 3.1.4 of the submittal, the licensee stated:

> 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 (HWL) 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.

To complete this action, operators will manually initiate a transfer of water inventory in the UCP and suppression pool known as the "drain-down" period, similar to the operation of the suppression pool makeup system. The water level requirements for the UCP and suppression pool ensures that post-accident water inventory is available, as required.

The licensee stated in its response to RAIs 1 and 2 that no new or changed operator actions are required as a result of the proposed installation of the reactor well to the steam dryer storage pool gate in the UCP in MODES 1, 2, and 3, nor will they be required as a result of the creation of proposed Special Operations TS 3.10.9, "Suppression Pool Makeup – MODE 3 Upper Containment Pool Drain-Down" (ADAMS Accession No. ML16356A005). Additionally, because there are no new or changed operator actions, verification and validation activities are not planned as stated in response to RAI 5. Furthermore, training for operators is not affected (ADAMS Accession No. ML16356A005). Based on the above, the NRC staff finds the description of operator actions to be acceptable.

3.7.2 Procedures

As stated above, there are no new or changed operator actions; therefore, there are no procedure changes related to operator actions. In the letter dated December 20, 2016, in response to RAI 1, the licensee stated that the new applicable procedures associated with the proposed installation of the reactor well to steam dryer pool gate in the UCP in MODES 1, 2, and 3, and as a result of the proposed creation of TS 3.10.9 will not be finalized until the implementation period following issuance of the license amendment (ADAMS Accession No. ML16356A005). The licensee further stated that "the procedures will be approved and made effective, concurrent with the implementation date of the license amendment." Additionally, there will be no changes to the emergency operating procedures or the off-normal instructions. Based on the above, the staff finds this position to be acceptable.

3.8 Evaluation of Proposed Changes to Technical Specifications

The NRC staff reviewed the technical discussion of the proposed changes provided in the LAR to ensure the reasoning was logical, complete and clearly written as described in Chapter 16 of NUREG-0800. The staff reviewed the proposed changes for continued compliance with the regulations in 10 CFR 50.36 and for consistency with conventional terminology and with the format and usage rules embodied in the TS.

The NRC staff reviewed the proposed changes to TS LCO 3.6.2.2 and conforming changes to SRs 3.6.2.4.1 and 3.6.2.4.4. The changes to LCO 3.6.2.2 are acceptable because the LCO continues to specify the lowest functional capability or performance levels of equipment required for safe operation.

The changes to the SRs for the SPMU are appropriate because the SRs, as revised, provide verification that the SPMU trains are operable and reflect the revised levels specified in LCO 3.6.2.2. The changes to SR 3.6.2.4.1 are necessary to ensure the UCP water level is appropriate depending on whether the reactor well to steam dryer storage pool gate is installed. The change to SR 3.6.2.4.4 is necessary to clarify that the SR is not applicable when the UCP levels are maintained per SR 3.6.2.4.1.c, no work is being done with the potential to drain the upper fuel transfer pool, the IFTS carriage is stored in the upper pool, and the IFTS transfer tube shutoff valve is closed.

Section 50.36(c)(2)(i) of 10 CFR requires that the TSs will include LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Section 50.36(c)(2)(i) of 10 CFR additionally requires that when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. No changes are being proposed to the remedial actions.

The NRC staff reviewed the proposed new LCO 3.10.9, Suppression Pool Makeup – MODE 3 Upper Containment Pool Drain Down. The LCO specifies the minimum requirements that must be met in MODE 3 with LCO 3.6.2.2 and 3.6.2.4 not met during drain down evolutions. The staff reviewed the associated ACTIONS Table. A separate condition entry is allowed for each Condition. Condition A requires that with one of the LCO requirements not met, that draining of the UCP be suspended immediately and restoration of compliance with the LCO requirements be restored within 4 hours. If Condition A is not met, Condition B requires restoration of

compliance with suspended MODE 3 LCO requirements within 12 hours. If Condition B is not met, Condition C requires entry into MODE 4 within 24 hours.

The SRs associated with LCO 3.10.9 include verification of suppression pool temperature, reactor steam dome pressure, levels in the Acceptable range of Figure 3.10.9-1, levels in the steam dryer storage pool and the fuel transfer pool areas of the UCP are sufficiently above the reactor vessel flange, location of the IFTS carriage, and closure of the IFTS transfer tube shutoff valve.

The bases for the LCO for suppression pool water level states that the LCO meets Criterion 1 and Criterion 3 of the Final Policy Statement for TSs (now codified in 10 CFR 50.36(c)(2)(ii)). The NRC staff determined LCO 3.10.9 would satisfy these same criteria during drain down evolutions when under the provisions of LCO 3.10.9.

Compliance with the requirements in LCO 3.10.9 is optional – i.e., compliance with the normal MODE 3 LCOs is an acceptable alternative. The NRC staff reviewed the proposed requirements and determined that the proposed requirements reflect the major assumptions supporting the engineering analyses discussed in this SE, and are, therefore, appropriate. The SRs associated with LCO 3.10.9 provide verification that the major assumptions of the supporting engineering analyses are continually met while relying on the provisions of this LCO.

For the reasons described above, the staff finds that revised TS continue to satisfy the requirements of 10 CFR 50.36(c)(2) and 50.36(c)(3).

The licensee provided a description of the proposed changes to the TS Bases in its initial application for information only. The licensee stated that the TS Bases are controlled by TS 5.5.11, "TS Bases Control Program."

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (81 FR 28898; dated May 10, 2016). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by

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

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Date of issuance: February 16, 2017.

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT REGARDING TECHNICAL SPECIFICATION CHANGES TO PERMIT INSTALLATION OF UPPER CONTAINMENT POOL GATE AND DRAIN-DOWN OF REACTOR CAVITY PORTION OF UPPER CONTAINMENT POOL (CAC NO. MF7476) (L-16-083) DATED

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