FENOC™
FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
P.O. Box 97
10 Center Road
Perry, Ohio 44081

David B. Hamilton
Vice President

440-280-5382

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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
Response to a Request for Additional Information Regarding a 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 (CAC No. MF7476)

By letter dated March 15, 2016 (Accession No. ML16075A411), FirstEnergy Nuclear Operating Company (FENOC) submitted to the Nuclear Regulatory Commission (NRC) a license amendment request for the Perry Nuclear Power Plant (PNPP). The proposed amendment would modify Technical Specification (TS) 3.6.2.2, "Suppression Pool Water Level," and TS 3.6.2.4, "Suppression Pool Makeup (SPMU) System," to allow installation of the reactor well to steam dryer storage pool gate 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 upper containment pool in MODE 3.

By letter dated October 6, 2016 (Accession No. ML16279A043), the NRC requested additional information to complete its review. FENOC's response to this request is attached.

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 November 7, 2016.

Sincerely,

David B. Hamilton

Attachment:
Response to Request for Additional Information

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

By letter dated March 15, 2016, FirstEnergy Nuclear Operating Company (FENOC) submitted a license amendment request for Nuclear Regulatory Commission (NRC) review and approval. By letter dated October 6, 2016, NRC staff requested additional information to complete its review. The requested information is presented in bold type, followed by the FENOC response.

1. **General Design Criteria (GDC) 16 requires that the reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Describe the containment spray system (CSS) actuation timing and summary of inputs used in calculating CSS set point. Confirm that the GOTHIC analyses performed in support of this LAR [license amendment request] do not impact the CSS set points described in Section 6 of the PNPP updated safety analysis report.**

Response:

The analytical limit for the containment spray initiation setpoint was specified by the General Electric design of the Perry Nuclear Power Plant (PNPP) as 23.7 pounds per square inch absolute (psia) or 9 pounds per square inch gauge (psig) [23.7 – 14.7 = 9]. The 9 psig value is cited in the PNPP Updated Safety Analysis Report (USAR), Section 6.2.1.1.5.4. The PNPP setpoint calculation methodology utilizes the General Electric, NEDC-31336, "Instrument Setpoint Methodology." The PNPP containment spray initiation setpoint analytical limit is 23.42 psia, which is based on the specific atmospheric pressure value at the PNPP of 14.42 psia. In determining the setpoint allowable value, the setpoint calculation considers loop accuracy, loop calibration, process measurement accuracy, primary element accuracy, and voltage drop. In determining the nominal trip setpoint, the methodology also factors in loop instrument drift. All but the voltage drop are combined by "square root sum of the squares," resulting in a calculated containment spray initiation setpoint allowable value of 8.715 psig. The containment pressure – high allowable value cited in Technical Specification 3.3.6.2, "RHR Containment Spray System Instrumentation,"
Table 3.3.6.2-1, is ≤ 8.71 psig, and is conservative to the calculated value.

The setpoint methodology also determines a nominal trip setpoint and a nominal trip setpoint for licensee event report (LER) avoidance by determining a standard deviation term associated with a 90 percent probability. Although the setpoint calculation provides for a nominal trip setpoint of 8.711 psig, the current plant setpoint is conservatively set to 8 psig. As stated in the March 15, 2016 letter, Section 3.7 of the Enclosure, the GOTHIC analysis initiated containment spray at 9 psig consistent with USAR Section 6.2.1.1.5.4.

In accordance with the General Electric design of the PNPP, under post-accident conditions, containment spray shall be operating at full spray flow within three minutes after containment spray initiation. Furthermore, ten minutes after a high drywell pressure signal, the containment spray will auto start in the presence of high containment pressure.

The General Electric design is reflected in PNPP USAR Section 6.2.1.1.5.4 and lists one of the assumptions for the allowable bypass calculations as, "Containment (spray) is activated 180 seconds after containment pressure reaches 9 psig or at LOCA plus 13 minutes, whichever occurs later."

As shown in the March 15, 2016 letter, Attachment 4, Figures 8 and 13 of the Enclosure, the GOTHIC analysis showed that containment spray initiated at approximately 780 seconds, which is ten minutes of high drywell pressure plus three minutes to achieve full containment spray flow.

Thus, the GOTHIC analysis for the proposed amendment does not alter the current containment spray initiation setpoint or timing as described in the PNPP USAR.

2. **GDC 50 requires that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. Describe the conservatisms used in the 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.**

Response:

The current licensing-basis steam bypass analysis for the PNPP is described in USAR Section 6.2.1.1.5, "Steam Bypass of the Suppression Pool;" USAR Table 6.2-3, "Engineered Safety Feature Systems Performance Parameters for Containment

Response Analyses;" USAR Table 6.2-10, "Available Containment Heat Sinks;" and USAR Figure 6.2-25, "Containment Pressure Following a Small Break with Steam Bypass (With Containment Spray and Heat Sinks & a minimum Mark III Design of $A/\sqrt{K} = 1.0$ ft^2)."

USAR Section 6.2.1.1.5.2, "Criteria," states, "The allowable bypass leakage is defined as the amount of steam which could bypass the suppression pool without exceeding the design containment pressure."

Thus, the current licensing basis steam bypass analysis determined the allowable drywell leakage capability such that the containment design pressure of 15 psig will not be exceeded. USAR Section 6.2.1.1.5.4 lists assumptions for the licensing basis steam bypass analysis. USAR Figure 6.2-25 provides the containment pressure response of a typical Mark III containment to a small steam break in the drywell and a steam bypass area $A/\sqrt{K} = 1.0$ square foot (ft^2).

To support the license amendment request for early drain down in MODE 3, the GOTHIC model was first used to simulate a small MODE 1 reactor pressure boundary steam break in the drywell with an assumed bypass of the suppression pool equal to the PNPP design bypass capacity of $A/\sqrt{K} = 1.68$ ft^2. This analysis was performed to validate that the GOTHIC model can predict the peak containment pressure at the maximum allowable bypass leakage of $A/\sqrt{K} = 1.68$ ft^2. Several MODE 1 benchmark runs were made, each varying the steam break size from 0.07 ft^2 to a maximum break size of 3.50 ft^2.

The GOTHIC model for LOCA with steam bypass included several assumptions that are similar to the licensing-basis methodology described in the PNPP USAR. These assumptions follow:

- The GOTHIC model uses the same heat sinks as used in the General Electric model (surface areas and volumes as shown in USAR Table 6.2-10). Material properties assumed are typical of concrete and carbon steel. The Uchida correlation for convective heat transfer is also used. The analysis assumes a single failure of one loop of containment spray. Only one containment spray loop is used at the design basis flow rate of 5250 gallons per minute (gpm). Containment spray flow is cooled by the residual heat removal (RHR) heat exchanger using an assumed maximum service water temperature of 85°Fahrenheit (°F) and an assumed heat exchanger performance value of 440 british thermal units (BTUs) per second per °F.

- Containment spray initiates no earlier than 13 minutes after a LOCA signal or 180 seconds after containment pressure exceeds 9 psig, whichever occurs later.
- A 100°F/hour reactor pressure vessel cooldown rate is assumed.

Several of the assumptions used in the GOTHIC model may be different than those used in the original licensing basis analysis. For example, stratification in the drywell is accomplished by subdividing the drywell into two vertical cells. The steam break is placed in the upper cell, so steam from the break will push air downward and out of the cell. Steam in this upper cell will then be at a higher temperature than in the lower cell. The bypass to the containment airspace is also placed in this upper cell. This results in bypass flow quickly becoming a higher temperature steam, which will conservatively pressurize the containment faster.

The GOTHIC model for LOCA with steam bypass also included several additional conservative assumptions that follow:

- 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 steam heating of the suppression pool as steam is directed to the suppression pool via both the vents and directly via the safety/relief valves (S/RVs). The heating of the suppression pool results in a higher water temperature at the inlet of the RHR heat exchanger, thus a higher temperature containment spray.
- The GOTHIC model includes heat from the operating containment spray pump, which results in a slightly higher temperature spray.

- The GOTHIC model includes modeling the reactor pressure vessel (RPV) with decay heat from the fuel and metal heat sinks. As the reactor steams, the liquid level in the reactor decreases. Level in the reactor is normally recovered using the high pressure core spray (HPCS) system drawing suction from the suppression pool. However, the GOTHIC model also includes the large volume of hot liquid captured in the feedwater piping, and the GOTHIC model prefers makeup using the feedwater liquid rather than HPCS system liquid until the feedwater volume is exhausted. This results in maximizing the heat, and thus, the steam source of the RPV.
- The GOTHIC model includes isolation of the RPV soon after the event initiates without any immediate pressure reduction. This causes reactor pressure to increase and, for small breaks, the opening of the S/RVs at their setpoints. When RPV makeup occurs, reactor pressure decreases slightly, and when makeup stops at the RPV high level (Level 8), operators wait until reactor pressure again increases to match the depressurization profile as they continue manually depressurizing the vessel.
- The GOTHIC model includes the effect of the suppression pool makeup system upper pool dump, which also increases the suppression pool level, adding back pressure on the drywell vents, increasing the differential pressure between drywell and containment, and increasing steam bypass flow.

The GOTHIC benchmark runs produce the same conclusion as the licensing basis analysis, that the PNPP containment design can withstand a postulated LOCA of any size with a steam bypass as large as $A/\sqrt{K} = 1.68\ ft^2$ and still maintain containment pressure below the design pressure of 15.0 psig. The benchmark run that resulted in the highest containment pressure was for a 0.50 ft^2 break.

For the benchmark steam bypass GOTHIC run with the limiting 0.50 ft^2 break size, the GOTHIC results are similar to USAR Figure 6.2-25, noting the two distinct peaks in containment pressure. The first peak occurs at about 780 seconds (13 minutes) when containment spray initiates. Containment pressure increases due to the bypass leakage, but when containment spray initiates, containment pressure is reduced. However, as spray continues, the spray becomes less effective and containment pressure again begins to increase. The second peak occurs more gradually as operators reduce reactor pressure below drywell pressure and the steam break flow decreases to a low value. Continued containment spray reduces the containment pressure and, as a result, the drywell pressure, and the event is over.

Thus, the GOTHIC model for LOCA with steam bypass is an acceptable model for re-creating the current licensing basis analysis. The GOTHIC model also has the capability to analyze the same event, but with conditions assumed for the early drain down in MODE 3.

The GOTHIC model for LOCA with steam bypass was then used for the steam bypass with early drain down in MODE 3 scenario. Conditions for the scenario include the reactor is shut down for two hours, in MODE 3, and steam dome pressure is no greater than 250 psia. The GOTHIC model for analysis at drain down conditions includes similar assumptions as before, with the following changes:

- Suppression pool level is assumed to be near the top of the weir wall with no initial differential pressure between drywell and containment. The pool level is intentionally higher than the analytical maximum level prior to early drain down to provide additional conservatism.
- RPV water level makeup is from feedwater liquid rather than HPCS liquid. A conservative feedwater enthalpy is assumed such that feedwater is initially injected at 350 BTUs per pound mass and reduces over time.
- Upper pool dump is included, but the volume available is assumed to be no greater than 6200 cubic feet.

Review of the containment response for the steam bypass with early drain down in MODE 3 analysis reveals that both pressure peaks remain below the containment design pressure with RPV pressure no greater than 250 psia. This confirms the requirement that the reactor pressure for early drain down in MODE 3 must be no greater than 250 psia.