FENOC
FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
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March 8, 2013
L-13-071

10 CFR 50.90

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

SUBJECT:
Perry Nuclear Power Plant
Docket Number 50-440, License Number NPF-58
Response to a Request for Additional Information on a License Amendment Request to Revise Technical Specification 3.10.1 (TAC No. ME9006)

By letter dated February 22, 2012 (Accession No. ML12054A061), the FirstEnergy Nuclear Operating Company requested Nuclear Regulatory Commission (NRC) approval to revise Perry Nuclear Power Plant (PNPP) Technical Specification 3.10.1 to expand its scope to include provisions for temperature excursions greater than 200 degrees Fahrenheit as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in Mode 4.

By letter dated February 7, 2013 (Accession No. ML13037A251), the NRC staff provided a request for additional information (RAI). The response to the RAI 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 March 8, 2013.

Sincerely,

Harlan D. Hanson, Jr.
Director, Site Performance Improvement

Attachment:
Response to Request for Additional Information on Technical Specification 3.10.1

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

By letter dated February 22, 2012 (Accession No. ML12054A061), the FirstEnergy Nuclear Operating Company (FENOC) requested Nuclear Regulatory Commission (NRC) approval to revise Perry Nuclear Power Plant (PNPP) Technical Specification (TS) 3.10.1 to expand its scope to include provisions for temperature excursions greater than 200 degrees Fahrenheit as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in Mode 4. By letter dated February 7, 2013 (Accession No. ML13037A251), the NRC staff requested additional information to complete its review. The NRC staff question is presented in bold type, followed by the FENOC response.

The model safety evaluation (SE) for the standard TS 3.10.1 specifically requires that secondary containment and standby gas treatment system shall be operable and capable of handling any airborne radioactivity or steam leaks that may occur during performance of testing. After the staff's review of PNPP's current TS 3.10.1, there appears to be a deviation from the Technical Specification Task Force (TSTF)-484 model SE.

1. **Please provide the technical and licensing bases for the exclusion of operability requirements in TS 3.10.1 for secondary containment and the annulus exhaust gas treatment system (the PNPP equivalent to the standby gas treatment system shown in the TSTF).**

Response:

Summary

The NUREG-1434 "Improved BWR-6 Technical Specifications," Revision 0, Technical Specification (TS) 3.10.1, "Inservice Leak and Hydrostatic Testing Operation," requires implementing secondary containment and standby gas treatment system requirements. In the mid-1990s, the PNPP TS were converted to the NUREG-1434 specifications. During the PNPP conversion, TS 3.10.1 was revised by replacing the secondary containment and standby gas treatment requirements with primary containment requirements. PNPP's conversion, including the revised TS 3.10.1, was approved by the NRC by Amendment 69 dated June 23, 1995. The basis of the TS 3.10.1 Limiting Condition for Operation (LCO) change is due to PNPP-specific design features such that the use of the primary containment is the more appropriate boundary for control of fission products.

Primary Containment

The primary containment is a pressure retaining structure composed of a free standing steel cylinder with an ellipsoidal dome, secured to a steel lined reinforced concrete foundation mat. The primary containment is surrounded by a reinforced concrete structure (the shield building). PNPP Updated Safety Analysis Report (USAR) Figure 3.8-1 provides a simplified cross-section of the two structures.

The primary containment is designed to contain radioactive material that might be released from the reactor coolant system following a loss-of-coolant accident (LOCA). The primary containment ensures a high degree of leak tightness during normal operating and accident conditions. It is designed for a maximum internal pressure of 15 pounds per square inch gauge (psig) at accident conditions. The maximum design leakage rate is 0.2 percent by weight of the contained atmosphere in 24 hours at 7.80 psig internal pressure.

There are two personnel airlocks that permit passage from the outside of the shield building into the primary containment. PNPP USAR Figure 3.8-4 provides a drawing of an airlock.

For equipment access into the primary containment during plant outages, there is a removable, bolted equipment hatch. PNPP USAR Figure 3.8-5 provides a drawing of the equipment hatch.

Mechanical and electrical lines that enter or exit the primary containment pass through penetrations located in both the primary containment and shield building walls. Typical mechanical and electrical penetrations are shown in PNPP USAR Figures 3.8-6 and 3.8-7.

In order to minimize the amount of primary containment leakage following a design basis accident, mechanical systems that pass through both the primary containment and shield building are typically provided with redundant containment isolation barriers, one inside of the primary containment and one outside of the shield building, or some other acceptable configuration such as a closed system outside of containment.

USAR Table 6.2-32, "Containment Isolation Valve Summary," provides a listing of the primary containment penetrations and their associated primary containment isolation barriers. The majority of the penetrations bypass the secondary containment, and any leakage would not be processed by the annulus exhaust gas treatment (AEGT) system. Specific leakage limits for these penetrations have been established and are consistent with the existing accident analysis. USAR Table 6.2-33, "Potential Secondary Containment Bypass Leakage Paths," provides the potential leakage paths that could bypass the secondary containment and the AEGT system.

Secondary Containment

The shield building is a reinforced concrete structure consisting of a flat foundation mat, a cylindrical wall and a shallow dome. The functions of the shield building include but are not limited to:

- Provides weather and exterior missile protection for the primary containment.
- Provides a relatively leak tight structure that collects fission product leakage from the primary containment during and following a postulated design basis accident and delays it until processing it through the annulus exhaust gas treatment (AEGT) system to minimize the escape of radioactive particles to the environment, by maintaining the annulus air space at a slight negative pressure.

The secondary containment at PNPP consists of the volume (or annulus) between the shield building and the primary containment. This area is approximately five feet wide, and has numerous systems that pass through both the primary containment and shield building walls as shown by PNPP USAR Figure 6.2-60 (Sheets 1 through 4). There are two doors, which are normally locked, that permit access to the annulus. Additionally, there is a large opening in the shield building to provide access to the primary containment equipment hatch. This opening is shielded by eight large removable reinforced concrete beams (called shield beams or shield blocks) as shown in PNPP USAR Figure 3.8-5, which are provided with seals to minimize air leakage. In order to establish secondary containment OPERABILITY, the activities include but are not limited to ensuring the primary containment equipment hatch and the annulus access doors are closed, ensuring the shield beams (with seals) are installed, and the AEGT system is OPERABLE.

The AEGT system functions to maintain a negative pressure differential between the annulus and outside of the shield building. The AEGT system processes the air in the annular space between the shield building and the primary containment to limit the release to the environment of radioisotopes that may leak from the primary containment under accident conditions. Sources of leakage into the annulus (secondary containment) include, but are not limited to, primary containment shell leakage and primary containment airlock outer door seal leakage. The AEGT system is a recirculation type system with split flow. Some of the filtered air extracted from the annulus space is recirculated, and some is discharged to the unit vent. The system has two subsystems; each subsystem includes a high efficiency particulate (HEPA) pre-filter, a charcoal adsorber, and a HEPA post-filter.

LOCA Analysis

The PNPP LOCA analysis considers a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and liquid process system lines. The initial conditions for the containment response aspect of this event include plant power at 3,833 megawatts-thermal, reactor vessel steam dome pressure at 1,060 pounds per square inch-absolute, and steam flow of approximately 16.7 million pounds mass per hour. The radiological aspect of the LOCA analysis includes postulating a fission product release to the containment.

The design basis of the primary containment after a LOCA event is to maintain its integrity such that the design limits are not exceeded. Since the primary containment design limit is 15 psig and the peak primary containment pressure during a design basis LOCA event does not reach this pressure, the design limit is not exceeded.

The radiological analysis of the LOCA event was revised as a result of the application of the revised accident source term at PNPP (Amendment 103). The dose calculations made a number of assumptions including but not limited to: no credit for iodine removal by the AEGT system charcoal adsorbers, increased engineered safety feature system leakage outside of containment, and increased secondary containment bypass leakage. However, the TS controls over the AEGT system charcoal adsorbers and the secondary containment bypass leakage were not requested to be changed as part of Amendment 103. The analysis considered four potential fission product release pathways:

- main steam isolation valve leakage,
- secondary containment bypass leakage,
- post-accident water leakage from engineered safety feature system leakage outside of containment, and
- containment leakage.

For the main steam lines, the fission product leakage is assumed to be released directly to the environment. Since there are systems that penetrate both the primary containment and shield building boundaries, the potential for pathways bypassing the collection and filtration systems of secondary containment is created. The analysis assumes that a percentage of primary containment leakage bypasses the secondary containment. TS provide a limit on this value. Leakage of water from engineered safety feature components located outside of the primary containment would release fission products outside of the secondary containment during the recirculation phase of long-term core cooling following the LOCA. Administrative controls limit this leakage to less than 50 percent of the value used in the radiological dose calculations. For containment leakage, the leakage is collected in the secondary containment for processing.

Note that three of the aforementioned release pathways bypass the secondary containment and the AEGT system. Though the fourth release pathway does enter the secondary containment, no credit is taken for iodine removal by the AEGT system charcoal adsorbers. The results of the radiological analysis indicate that the licensing basis limits are satisfied.

Evaluation

The conditions used in the LOCA analysis include the plant at high power, high reactor pressure, and high steam flow. This is in contrast with the plant conditions when TS 3.10.1 would be entered. The conditions for TS 3.10.1 is at the end of a refueling outage when the reactor vessel is nearly water solid, at relatively low decay heat values, in Mode 4 (Cold Shutdown), and the reactor core not critical. The relative stored energy in the reactor core during this period will be very low. Small reactor coolant system leaks would be detected by leakage inspections before significant inventory loss occurred. In the event of a large primary system leak, the reactor vessel would rapidly depressurize, allowing the low pressure core cooling systems to operate. The capability of the low pressure coolant injection and low pressure core spray subsystems would be more than adequate to keep the core flooded under this low decay heat load condition. Under these conditions, the potential for failed fuel and a subsequent increase in the reactor coolant activity above the TS values is minimized.

Though the subject leak/hydrostatic test does pressurize the reactor coolant system, the testing is performed when energy levels in the core are low. If a leak would occur during this testing, there should be insufficient energy to pressurize the containment and provide the driving force for leakage approaching the primary containment design value.

Therefore, implementing primary containment requirements in lieu of secondary containment and AEGT system requirements establishes more robust containment conditions. The primary containment is designed as a pressure retaining structure with a high degree of leak tightness. Primary containment barriers would be controlled in accordance with TS requirements. Leakage into the secondary containment through the containment shell and other paths is expected to be minimal due to the limited energy conditions postulated during the period that TS 3.10.1 would be used. This leakage could be processed by the fuel handling area ventilation system, which serves the building that is connected to the annulus (secondary containment) when the shield beams are not installed. The exhaust subsystem of the fuel handling area ventilation system contains three filter trains; the trains include HEPA pre-filters, charcoal filters, and HEPA after-filters.

Conclusion

In summary, the primary containment is designed to be a pressure retaining structure. During the period that TS 3.10.1 would be used, if a leak would occur during the leak/hydrostatic testing, there would be insufficient energy to pressurize the containment and provide a driving force for leakage approaching the primary containment design value. Plant conditions during this period are such that if a reactor coolant system leak were to occur, the probability of core damage and a subsequent increase in the reactor coolant activity above the TS values is low. Primary containment barriers would be controlled in accordance with TS requirements, and consequences would be bounded by the existing accident analysis. With limited energy conditions expected while under the provisions of TS 3.10.1, leakage into the secondary containment would be limited, and could be processed by the fuel handling area ventilation system. Additionally, existing accident analysis does not credit AEGT system charcoal adsorbers for iodine removal.