Entergy

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Pilgrim Nuclear Power Station
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Stephen J. Bethay
Director, Nuclear Assessment

March 13, 2007

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

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293 License No. DPR-35

License Renewal Application Amendment 15

REFERENCE: Entergy letter, License Renewal Application, dated January 25, 2006 (2.06.003)

LETTER NUMBER: 2.07.025

Dear Sir or Madam:

In the referenced letter, Entergy Nuclear Operations, Inc. applied for renewal of the Pilgrim Station operating license. NRC TAC NO. MC9669 was assigned to the application.

This License Renewal Application (LRA) amendment consists of Attachment A which contains additional information in response to the request for additional information (RAI) on LRA Section B.1.16.1 Containment Inservice Inspection, conveyed in NRC letter dated November 7, 2006.

There are no new commitments contained in this letter.

Please contact Mr. Bryan Ford, (508) 830-8403, if you have any questions regarding this subject.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 13, 2007.

Sincerely,

Stephen J. Bethay

Stephen J. Bethay
Director Nuclear Safety Assessment

ERS
Attachments: (as stated)
cc: see next page

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cc: with Attachments

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ATTACHMENT A to Letter 2.07.025

(8 pages)

Additional Information in Response to Request for Additional Information on LRA Section B.1.16.1 Containment Inservice Inspection

B.1.16.1 Containment Inservice Inspection (CII)

RAI B.1.16.1:

1. In the Pilgrim Nuclear Power Station (PNPS) aging management program B.1.16.1 of the license renewal application (LRA), the applicant stated: "CII inspections during RFO 15 (April 2005) did not reveal evidence of loss of material. Absence of degradation provides evidence that the program is effective for managing the aging effect." In addition, in the LRA, Amendment 2 (ML061710422) under, "Ongoing Actions to Prevent Drywell Corrosion," PNPS stated, "Functional checks are performed each refueling outage on the flow switch associated with the bellows seal leakage monitoring system." However, the recent NRC Region I inspection team observations indicated that:

   a. The flow switch in the bellows rupture drain had failed its surveillance in December 2005, and has not been fixed or evaluated. In addition, the flow switch also had been failed in 1999.

   b. Monitoring of other drains has been inconclusive and not well documented.

   c. The torus room floor has had water on the floor on multiple occasions.

   Please provide a detailed discussion, including record, corrective actions taken, and preventive action in response to this plant specific operating experience and discuss its impact on the aging management of potential loss of material due to corrosion in the inaccessible area of the Mark I steel containment drywell shell, including the sand pocket region for the period of extended operation.

Response to RAI B.1.16.1:

Entergy via License Renewal Application (LRA) Amendment 10 dated December 12, 2006 initially responded to this RAI. Additionally, LRA Amendment 13 dated January 29, 2007 provided information concerning drywell monitoring. Subsequent to these responses, discussions with the staff identified that additional information concerning the potential implications of the water seepage on the torus room floor was needed. This letter updates our response to provide this additional information.

a. On December 28, 2005, the flow switch in the bellows rupture drain (FS-4803) failed to respond acceptably during testing. During the test, water is poured into an upstream test funnel. The water normally flows into the flow switch, actuates the switch, and discharges to the radwaste system. On this occasion, the flow switch did not alarm. This was caused by blockage of the passages around the perimeter of the baffle of the flow switch. The apparent cause of the blockage was accumulation of crud and corrosion products from the test funnel and associated piping during routine testing.

   <u>Corrective actions</u>

   The flow switch associated with the bellows rupture drain was replaced with a new switch on November 17, 2006 and the drain functionally tested. The flow switch indicates rupture of the refueling bellows seal when the refueling cavity is full of water during refueling operations. The last time the cavity was filled was during the refueling outage ending in

May 2005. PNPS operates on a two-year refueling cycle, hence the next time the refueling cavity will be filled is in the spring of 2007. During the period from discovery of the FS-4803 failure to respond until replacement with the new switch in November 2006, there was no potential for undetected leakage since no water was present above the refueling bellows seal.

Preventive actions

A preventive maintenance task was established to replace flow switch FS-4803 every 15 years.

Functional checks of the flow switch in the bellows rupture drain (FS-4803) are performed prior to each refueling outage and the switch is repaired, if necessary.

Impact on management of loss of material due to corrosion in the inaccessible areas of the containment drywell shell

If leakage occurred with the flow switch in the bellows rupture drain (FS-4803) failed, leakage would enter the 8" casings where it would be indicated by leakage from the four ¾-inch refueling bellows rupture tell-tale drains. Daily operator rounds have not detected leakage from these tell-tale drains.

If FS-4803 fails to indicate leakage and leakage is not detected from the ¾-inch refueling bellows rupture tell-tale drains before the 8" casings fill up and water rises above the 4" high plate that surrounds the ledge to the air gap, leakage can overflow into the annulus air gap. A sheet metal cover plate shields the sand cushion area against leakage from above. Four 4" annulus air gap drain lines direct water from above the sand pocket to the torus room floor. These drains are checked by inservice inspection (ISI) VT-2 certified inspectors for leakage twice every refuel outage, once after refuel cavity flood up and again prior to draining down. No leakage has ever been detected from these drains. Containments placed under these annulus air gap drains in the 1980's are dry with no evidence of previous water accumulation.

The PNPS design features discussed above provide a robust barrier against undetected leakage into the sand cushion region. Since leakage has not been detected by any of the means that are part of this robust design, the temporary failure of FS-4803 had no impact on the aging management of the inaccessible areas of the Mark I steel containment drywell shell.

b. PNPS has four layers of refueling bellows leakage detection and water removal. LRA Amendment 13 dated January 29, 2007 provides additional description of the drains and monitoring systems. The four layers are the following:

- The instrumented drain line discussed in response "a" above drains a 2" drain trough and is functionally tested once per cycle. Corrective actions are taken as needed.

- If the 2" drain trough overflows and the leakage is not identified by the instrumented line, leakage is directed by a 4" high plate to four ¾-inch tell-tale drains which drain the bellows area. Surveillance of these four refueling bellows rupture tell-tale drains on the reactor building 74 ft. elevation is performed and documented twice daily during reactor building operator rounds during refueling. Visual observation of the tell-tale drains during these operator rounds has detected no leakage in the past.

- If the 2" drain trough overflows without the leakage being identified by the instrumented line and the 4" deep leakage direction plate is breached without the leakage being identified coming from the four ¾-inch drains, leakage would enter the annulus air gap region and be directed to four 4" annulus air gap drains by a plate which covers the top of the sand cushion region. These four 4" annulus air gap drains drain the area immediately above the sand cushion.

  The four 4" annulus air gap drains are located in bays 2, 6, 10 and 14 of the torus room adjacent to the reactor pedestal and discharge approximately 6" above the floor (containments were placed in the 1980s on the floor under the drains). The four drains are examined for leakage twice during each refueling outage as augmented examinations under the station's IWE containment inspection program. They are checked for leakage by ISI VT-2 certified examiners during outages once after flooding the refueling cavity and again prior to draining the refueling cavity at the close of the outage. Leakage, if present, would discharge into the containments on the floor of the torus room as the drain lines are not directed to floor drains. Leakage from the four annulus air gap drains (see response to c.) could not be mistaken for groundwater intrusion in the torus room, because leakage would be collected in the containments under the drains.

  PNPS has verified the annulus air gap drain lines are unobstructed. In 1982, the air lines were verified not to contain standing water. Subsequently, in 1987, access holes were machined in the drain line elbows on all four drain lines to allow access for remote visual examination using fiberscopes. This examination determined that the four annulus air gap drains are unobstructed and found no signs of corrosion on visible portions of the drywell surface.

  The containments under the annulus air gap drains were checked following the NRC Region 1 LRA inspection and no evidence of leakage from the drain lines was identified (i.e., the containments were in place, dry, and showed no evidence of having previously contained water).

- If the 2" drain trough overflows without the leakage being identified by the instrumented drain line, the 4" deep leakage direction plate is breached without the leakage being identified coming from the four ¾-inch drains, and leakage enters the sand cushion past the sealing plate without the leakage being identified by the four annulus air gap drains, the water would be directed to the four sand cushion drains. These four drains are filled with sand by design and drain into the torus room compartment about 12 ft. above the floor level. If these drains were draining water from the sand cushion region the drainage would be identified by personnel as water coming from the overhead and could not be mistaken for groundwater intrusion in the torus room.

<u>Corrective actions</u>

No additional actions are necessary.

<u>Preventive actions</u>

No additional actions are necessary.

Impact on management of loss of material due to corrosion in the inaccessible areas of the containment drywell shell

Since leakage has not been detected at any of the drains, there is no impact on management of loss of material due to corrosion in the inaccessible areas of the containment drywell shell for the period of extended operation.

c. In September 2006, shallow, puddled standing water was observed in torus bays 6, 7, 10 and 13. Although dry, the appearance of the floor in bay 11 indicated it had been wet. Dampness and corrosion was evident around the Williams Rock Anchor baseplates at torus saddles 6, 10, 11, 12, 13, 14 & 15. There was no condensation on the torus itself, nor any evidence of process system leakage.

The locations and extent of the observed conditions are consistent with conditions investigated and evaluated previously. At that time, a thorough investigation concluded that the cause of water on the torus room floor was groundwater infiltration by-passing the membrane system that encapsulates the reactor building foundation. A test performed in 1996 demonstrated that water was emerging from the anchor bolt holes for the Williams Rock Anchors at the torus saddles. Containments under the air gap annulus drains that continue to remain dry ensure that those drains are not contributing to the water observed on the floor.

Corrective actions

None.

Preventive actions

An assessment of the torus saddle anchor bolts in 1999 concluded that active corrosion of the embedded anchor bolts was not occurring. These supports and associated bolting are inspected under the Containment Inservice Inspection (CII) Program to assure that effects of aging will be managed such that they will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

The feasibility of pressure grouting the torus room floor or correcting the condition of floor surface drainage to allow water to flow to the floor drains is being evaluated.

Impact on management of loss of material due to corrosion in the inaccessible areas of the containment drywell shell

Since the water on the torus room floor is not attributed to refueling cavity leakage or leakage into the annulus air gap, this condition has no impact on management of loss of material due to corrosion in the inaccessible areas of the containment drywell shell for the period of extended operation. Collection containments under the annulus air gap drains continue to remain dry.

Other Considerations

Drywell Shell Corrosion

The condition of the sealant between the shield at the base of the annulus air gap and the drywell shell is uncertain because of its inaccessibility. Also due to inaccessibility, the absence of residual moisture in the sand cushion following leakage into the annulus air gap cannot be confirmed. For these reasons, augmented inspection of the drywell shell in the sand cushion area is warranted following indications of leakage into the annulus air gap.

Drywell Corrosion Monitoring Enhancements

If leakage into the inaccessible area adjacent to the exterior of the drywell shell (the annulus air gap) is detected or suspected, PNPS will document and evaluate the condition in accordance with the corrective action program. Appropriate corrective actions to correct the cause of the leakage and to address the potential degradation of the drywell shell caused by the condition would be initiated. To assure corrective actions include augmented inspections performed in accordance with ASME Section XI IWE-1240, the Containment Inservice Inspection (CII) Program is enhanced to require augmented inspection of the drywell shell adjacent to the sand cushion following indications of water leakage into the annulus air gap.

This enhancement, License Renewal Commitment 41, requires the following revisions to the LRA:

LRA Section B.1.16.1, Containment Inservice Inspection (CII), is revised to include the following.

The following enhancement will be implemented prior to the period of extended operation.

Attributes Affected	Enhancements
3. Parameters monitored and inspected	Administrative controls are revised to require augmented inspection in accordance with ASME Section XI IWE-1240 of the drywell shell adjacent to the sand cushion following indications of water leakage into the annulus air gap.

LRA Section A.2.1.17, Inservice Inspection – Containment Inservice Inspection (CII) Program, is revised to include the following statement.

"License Renewal Commitment 41 specifies an enhancement to this program."

In addition to the above enhancement, to provide added assurance that the actions to prevent water intrusion into the annulus air gap and sand cushion region have been successful and thereby, have mitigated the potential for corrosion, LRA Commitment 44 is to perform another set of the UT measurements at the previous measurement locations just above and adjacent to the sand cushion region prior to the period of extended operation and once within the first 10 years of the period of extended operation.

Commitment 44 will be reflected in FSAR supplement by adding the following to the second paragraph of LRA Section A.2.1.17.

"Prior to the period of extended operation, ultrasonic testing will be used to determine drywell shell thickness at previous measurement locations just above and adjacent to the sand cushion region. An additional set of measurements at the same locations will be performed once prior to the end of the first 10 years of the period of extended operation. License renewal commitment 44 specifies this additional testing."

Reactor Building Base Mat

The reactor building is founded at elevation -25.5 ft. msl on dense to very dense silty sand and sand and gravel. The top of bedrock is at about elevation -68.5 ft. msl and the ground water table is typically at about -1 ft. msl. The base mat of the reactor building is about 142 ft. 6 in. x 142 ft. 6 in. x 8 ft. thick, and the top of this mat is the floor of the torus room. It was constructed in five individual 8 ft. thick concrete pours, separated by vertical construction joints to minimize shrinkage cracking. The central portion of the base mat supports a 25 ft. thick reinforced concrete pedestal, which serves as the foundation for the drywell and reactor vessel. Groundwater in-leakage is designed to be prevented or minimized by a waterproof membrane that encapsulates the reactor building foundation.

The water on the torus room floor is believed to be the result of groundwater intrusion through imperfections in the encapsulating membrane. Once groundwater has penetrated the membrane, the most probable method of transport through the 8 ft. thick reinforced concrete base mat is hydrostatic pressure through the foundation construction joints. The observed field conditions include shallow, puddled water and indications of previous shallow, puddled water associated with embedded anchors near construction joints in some torus room bays. The extent of the water is limited, and because its ingress is in equilibrium with evaporation losses, pumping for removal is not required. Since the groundwater is not chemically aggressive to the concrete or embedded carbon steel, and since seepage of this nature in the absence of other symptoms is not believed to be indicative of a structural deficiency under service loadings, this condition is not a concern with regards to the structural capability of the reactor building base mat or embedded steel.

Reinforced concrete design for Class I buildings is in accordance with the requirements of American Concrete Institute Standard ACI 318-63 as described in FSAR Section 12.2.2.12. Code design practices for proportioning reinforcing steel are fundamentally based on the fact that concrete cannot resist significant tension and may crack due to shrinkage stresses during curing, or due to in-service stresses from temperature changes and structural loading. Design practice assumes that concrete subjected to tension will crack, and reinforcing steel is provided to ensure structural integrity for service loads under these conditions. Inspections of the reactor building base mat have not identified any cracking suggesting concerns with the structural capability of the structure. No cracks in the concrete are evident other than normal hairline shrinkage cracks.

The condition of the structure is monitored for evidence of damage or degradation on an ongoing basis by the maintenance rule structures monitoring program.

Reactor Building Foundation Conditions

During the construction of PNPS, in addition to the overall site geologic and seismic explorations, detailed foundation investigations, including borings and field permeability tests, were carried out for use in establishing the foundation criteria for the station structures. Prior to construction, a series of test borings were drilled in the general station area to determine subsurface conditions. Borings were made to various depths from 16 to 130 ft. and bedrock was generally encountered at a depth of about -68.5 ft. in the station area. Disturbed and undisturbed soil samples, suitable for laboratory testing, were extracted from the test borings, examined, and subjected to the laboratory tests listed in FSAR Section 12.2.4.3. FSAR Figures 12.2-6 through 12.2-10 show some of the boring logs for borings taken in the station area.

Based on the results of these investigations, the total settlement of the reactor building was estimated to be in the range of 2" to 4" of uniform elastic compression at the design loads, with differential settlement expected to be less than 1". This is discussed in FSAR Section 12.2.4.4.3. Since this structure consists primarily of dead load, some settlement occurred during the construction phase, and post construction settlement was expected to be on the order of ½-inch. A survey traverse of points on the building was established to monitor foundation performance during construction (FSAR Figure 12.2-11). FSAR Table 12.2-6 summarizes the results of foundation settlement measurements taken at various stages of dead load application during construction. No evidence has been observed that would indicate any additional structural settlement of the reactor building after 35 years of operation.

The conclusion that reactor building settlement is not of concern for the period of extended operation is consistent with the conclusions of EPRI report Aging Effects for Structures and Structural Components (Structural Tools), Revision 1 that concluded the following.

> "For concrete structures founded on dense soil, backfill, or partially weathered rock, if in the past 20 years of experience for a structure, the total differential settlement experienced are well within the permissible limits for this type of structure (i.e., tank foundation, pump house, turbine building) and no settlement has manifested itself via cracked walls or cracked foundations, then it can be concluded that cracking due to settlement is not significant and would not be applicable for the structure during the period of extended operation."

The condition of the reactor building is monitored for the evidence of abnormal settlement by the maintenance rule structures monitoring program.

Groundwater Impact

The ground water at PNPS is non-aggressive to the base mat. PNPS has sampled ground water at a well near the reactor building on two recent occasions. The results are as follows.

Date	Chlorides (ppm)	Sulfate (ppm)	pH
11/27/05	420	16	6.2
6/13/06	210	<5	6.3

In addition, the EPRI report titled Aging Effects for Structures and Structural Components (Structural Tools), Revision 1, Section 5 addresses concrete structures and concrete components. Section 5.3 states that test results show that the compressive strength of concrete actually increases with age and the high alkalinity of concrete protects embedded steel from corrosion. Further, the report states "if the concentrations of acid and aggressive chemicals are all below the threshold limits, aggressive chemicals is not an applicable aging mechanism for concrete structures and structural members".

The threshold levels for non-aggressiveness to embedded steel are the following.

pH > 5.5
Chlorides < 500 ppm
Sulfates < 1500 ppm

As demonstrated by the PNPS sampling, the groundwater at PNPS is not aggressive to the embedded structural steel of the base mat. To monitor the groundwater in the future, PNPS has made LRA Commitment 43 to include within the Structures Monitoring Program provisions to ensure groundwater samples are evaluated to periodically to assess the aggressiveness of groundwater to concrete, as described in Attachment E of LRA Amendment 12 (Letter 2.07.005). Commitment 43 will be implemented prior to the period of extended operation.

In general, ground water is not aggressive to metal embedded in concrete due to the pH of the water being increased as it passes through the concrete. To determine the aggressiveness of water infiltrating into the torus room, PNPS sampled the water collecting on the torus room floor. The following are the results.

March 1989	April 1999	February 2007
Chlorides - 120 ppm Nitrates - 20 ppm pH - 8.76 Calcium - 292 ppm	pH - 9.5	pH - 9.45

Note: The water pH would actually be greater than measured because the carbonization process lowers the pH of water exposed to air during sampling.

As demonstrated by the testing of the water in the torus room, the water is not aggressive to the metal embedded in concrete of the base mat. For additional assurance that the water will remain non-aggressive, PNPS has committed to additional sampling (Commitment 45). LRA Commitment 45 reads as follows.

"If groundwater continues to collect on the torus room floor, obtain samples and test such water to determine its pH and verify the water is non-aggressive as defined in NUREG-1801 Section III.A1 item III.A.1-4 once prior to the period of extended operation."