Entergy

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Stephen J. Bethay
Director, Nuclear Safety Assurance

January 31, 2011

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

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

Pilgrim Nuclear Power Station (PNPS) License Renewal Application (LRA) Additional Supplemental Information

REFERENCES:

1. Entergy Letter No. 2.06.003, to USNRC, "Entergy Nuclear Operations Inc., License No. DPR-35, License Renewal Application," dated January 25, 2006.

2. Entergy Letter No. 2.11.001, to USNRC, "Pilgrim Nuclear Power Station (PNPS) License Renewal Application (LRA) Supplemental Information," dated January 7, 2011.

LETTER NUMBER: 2.11.008

Dear Sir or Madam:

On January 25, 2006, Entergy Nuclear Operations, Inc. (Entergy) submitted the License Renewal Application (LRA) for the Pilgrim Nuclear Power Station (PNPS) as indicated by Reference 1.

On January 7, 2011, Entergy submitted supplemental information regarding five (5) areas related to the LRA for the Pilgrim Nuclear Power Station (PNPS) as indicated by Reference 2. These are; aging management of neutron-absorbing materials, inspection of socket welds in small-bore piping, inspection of buried pipe and tanks, aging management of low voltage cables, and inspection of containment coatings.

This letter provides additional supplemental information to the LRA to address the following four areas which Entergy agreed to evaluate and supplement the LRA, as necessary.

1. Environmentally Assisted Fatigue Analysis
2. One-Time Inspection Program
3. Selective Leaching of Materials Program
4. Structures Monitoring Program

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NRR

A new regulatory commitment is provided in Attachment 2.

Should you have any questions or require additional information concerning this submittal, please contact Mr. Joseph R. Lynch at 508-830-8403.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 31, 2011.

Sincerely,

Stephen J. Bethay

Stephen J. Bethay
Director Nuclear Safety Assessment

JRL/jl

Attachments: 1. License Renewal Application Additional Supplemental Information (9 Pages)
2. License Renewal Commitment List (2 Pages)

cc:

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Attachment 1 to Letter No. 2.11.008

Pilgrim Nuclear Power Station
License No. DPR-35 (Docket No. 50-293)

License Renewal Application

Additional Supplemental Information

Pilgrim Nuclear Power Station
License Renewal Application - Supplemental Information

Entergy provides the following supplemental information in response to NRC Draft Requests for Additional Information (RAI) and as a result of industry activities potentially relevant to aging management in the following areas at Pilgrim Nuclear Power Station (PNPS).

- Draft RAI 4.3.3-1 – Metal Fatigue NUREG/CR-6260
- Draft RAI B.1.23-1 - One-Time Inspection
- Draft RAI B.1.27-1 - Selective Leaching of Materials
- Draft RAI B.1.29.2-2 – Structures Monitoring Program Acceptance Criteria

DRAFT RAI 4.3.3-1 – Metal Fatigue NUREG/CR-6260

Background

In LRA Section 4.3.3 and Commitment No. 31, the applicant discussed the methodology used to determine the locations that required environmentally assisted fatigue analyses, consistent with NUREG/CR-6260 "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear power Plant Components." The staff recognized that, in LRA Table 4.3-3, there are nine plant-specific components listed, based on the six generic locations identified in NUREG/CR-6260. The first part of Commitment No. 31 indicated:

> At least 2 years prior to entering the period of extended operation, for the location identified in NUREG/CR-6260 for BWRs of the PNPS vintage, PNPS will refine our current analyses to include the effects of reactor water environment and verify that the cumulative usage factors (CUFs) are less than 1. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following:
>
> 1. For locations, including NUREG/CR6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF.
> 2. More limiting PNPS-specific locations with a valid CUF may be added in addition to the NUREG/CR-6260 locations.
> 3. Representative CUF values from other plants, adjusted to or enveloping the PNPS plant specific loads may be used if demonstrated applicable to PNPS.
> 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

The GALL Report AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary" states the impact of the reactor coolant environment on a sample of critical components should include the locations identified in NUREG/CR-6260, as a minimum, and that additional locations may be needed.

Issue

The staff identified two concerns regarding the applicant's environmentally assisted fatigue analysis and Commitment No. 31.

1. Item 2 in Commitment No. 31 indicated that more limiting plant-specific locations *may* be added. However, it is only one of the *options* that may be taken. The applicant has not committed to verify that the PNPS-specific components per NUREG/CR 6260 are bounding for the generic NUREG/CR-6260 locations in Commitment No. 31.

2. The staff noted that the applicant's plant-specific configuration may contain locations that should be analyzed for the effects of reactor coolant environment, other than those generic locations identified in NUREG/CR-6260. This may include components that are limiting or bounding for a particular plant-specific configuration or that have calculated CUF values that are greater when compared to the locations identified in NUREG/CR-6260. The staff noted that LRA Section 4.3.3 and Commitment No. 31 do not address this issue.

Request

1. Confirm and justify that the plant-specific components listed in LRA Table 4.3-3 are bounding for the generic NUREG/CR-6260 locations.

2. Confirm and justify that the LRA Table 4.3-3 components selected for environmentally assisted fatigue analyses consists of the most limiting component *for the plant* (beyond the generic locations identified in the NUREG/CR-6260 guidance). If these components are not bounding, clarify the components that require an environmentally assisted fatigue analysis and the actions that will be taken for these additional components. If the limiting component identified consists of nickel alloy, clarify that the methodology used to perform environmentally-assisted fatigue calculation for nickel alloy is consistent with NUREG/CR-6909. If not, justify the method chosen.

Response

Entergy will review design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the Pilgrim plant configuration. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage.

PNPS will use the NUREG/CR-6909 methodology in the evaluation of the limiting locations consisting of nickel alloy, if any. This evaluation will be completed prior to the period of extended operation.

DRAFT RAI B.1.23-1 – One-Time Inspection

Background

GALL AMP XI.M32, "One-Time Inspection" states in element 4, "detection of aging effects," that the inspection includes a representative sample of the system population, and, where practical, focuses on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin.

LRA Section B.1.23, One-Time Inspection Program stated that the program includes determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience; and identification of the inspection locations in the system or component based on the aging effect.

Issue

Due to the uncertainty in determining the most susceptible locations and the potential for aging to occur in other locations, the staff noted that large sample sizes (at least 20%) may be required in order to adequately confirm an aging effect is not occurring. The applicant's One-Time Inspection Program did not include specific information regarding how the population of components to be sampled or the sample size will be determined.

Request

Provide specific information regarding how the population of components to be sampled will be determined and the size of the sample of components that will be inspected.

Response

License renewal application (LRA) Section B.1.23, One-Time Inspection, describes the determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience and identification of the inspection locations in the system or component based on the aging effect. PNPS is providing additional specific information regarding the sample of components that will be inspected.

Representative samples are chosen from each population. Each group of components with the same material-environment combination is considered a separate population. The sample size is based on Chapter 4 of EPRI Report TR-107514, "Age Related Degradation Inspection Method and Demonstration," which outlines a method to determine the number of inspections required for 90% confidence that 90% of the population does not experience degradation (90/90). Inspection locations are determined based on susceptibility, accessibility, dose considerations and operating experience. Where practical, inspections focus on the bounding or lead components most susceptible to aging due to time in service and severity of operating conditions.

For small populations (100 or less), the EPRI TR-107514 criterion will be modified such that the sample is at least 20% of the population with no less than 2 inspections. This method provides a reasonable sample number for all populations.

The PNPS sampling approach provides confirmation of the effectiveness of aging management programs in assuring that systems will remain capable of performing their intended functions through the period of extended operation.

DRAFT RAI B.1.27-1 – Selective Leaching of Materials

Background

GALL AMP XI.M33, “Selective Leaching of Materials” states in element 1, “scope of program” that the program includes a one-time visual inspection and hardness measurement of a selected set of sample components to determine whether loss of material due to selective leaching is not occurring for the period of extended operation.

LRA Section B.1.27, Selective Leaching Program, stated that the program will include a one-time visual inspection and hardness measurement of selected components that may be susceptible to selective leaching to determine whether loss of material due to selective leaching is occurring.

Issue

Due to the uncertainty in determining the most susceptible locations and the potential for aging to occur in other locations, the staff noted that large sample sizes (at least 20%) may be required in order to adequately confirm an aging effect is not occurring. The applicant's Selective Leaching Program did not include specific information regarding how the selected set of components to be sampled or the sample size will be determined.

Request

Provide specific information regarding how the selected set of components to be sampled will be determined and the size of the sample of components that will be inspected.

Response

License renewal application (LRA) Section B.1.27, Selective Leaching, describes the determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience and identification of the inspection locations in the system or component based on the aging effect. PNPS is providing additional specific information regarding the sample of components that will be inspected.

Representative samples are chosen from each population. Each group of components with the same material-environment combination is considered a separate population. The sample size is based on Chapter 4 of EPRI Report TR-107514, “Age Related Degradation Inspection Method and Demonstration,” which outlines a method to determine the number of inspections required for 90% confidence that 90% of the population does not experience degradation (90/90). Inspection locations are determined based on susceptibility, accessibility, dose considerations and operating experience. Where practical, inspections focus on the bounding or lead components most susceptible to aging due to time in service and severity of operating conditions.

For small populations (100 or less), the EPRI TR-107514 criterion will be modified such that the sample is at least 20% of the population with no less than 2 inspections. This method provides a reasonable sample number for all populations.

The PNPS sampling approach provides reasonable assurance that components potentially susceptible to selective leaching will remain capable of performing their intended functions through the period of extended operation.

Draft RAI B.1.29.2-2: Structures Monitoring Program Acceptance Criteria

Background

The GALL Report AMP IX.S6 "Structures Monitoring Program," states that American Concrete Institute (ACI) Section 349.3R is an acceptable basis for selection of parameters monitored, detection of aging effects (i.e. inspection interval), and acceptance criteria. Recent staff reviews have identified license renewal applications state that the Structures Monitoring Program is comparable to the GALL Report AMP; however the staff found the applicant's actual acceptance criteria is less conservative than the recommendations in the GALL Report AMP.

Issue

The LRA did not clearly identify quantitative acceptance criteria for Structures Monitoring Program inspections.

Request

a) Confirm that the quantitative acceptance criteria for the Structures Monitoring Program is consistent with the criteria of ACI 349.3R. If the criteria deviate from those discussed in ACI 349.3R, provide technical justification for the differences.
b) If quantitative acceptance criteria will be added to the program as an enhancement, provide plans and a schedule to conduct a baseline inspection with the quantitative acceptance criteria prior to the period of extended operation.

Response

(Part a)
For concrete structures, the Structures Monitoring Program (SMP) has a responsible engineer with the appropriate education and experience to identify and evaluate existing conditions using the appropriate industry standards for concrete structures, including ACI standards. The SMP will be enhanced to include more detailed guidance on quantitative acceptance criteria of ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" for concrete structures prior to the period of extended operation (PEO).

Commitment

Entergy is providing the following commitment (Commitment 51) for the Structures Monitoring Program;

Enhance the Structures Monitoring Program to invoke quantitative acceptance criteria for inspections of concrete structures in accordance with ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" prior to the period of extended operation (PEO).

(Part b)
Program procedures specify that the inspection engineer be a degreed engineer or registered professional engineer, knowledgeable or trained in the design, evaluation, and performance requirements of structures, with at least 5 years structural design/analysis/field evaluation experience. Using applicable industry codes and standards, the responsible engineer has adequate training and education to determine the acceptability of identified conditions using appropriate references, which may include ACI 349.3R.

While all the detailed quantitative acceptance criteria of ACI 349.3R are not in the existing SMP procedures, the knowledge and experience of the qualified inspection engineers performing regularly scheduled inspections provides reasonable assurance of continued functionality of the concrete structures at PNPS. The enhanced inspection criteria from ACI 349.9-3R will be adopted prior to the PEO and will be applied during regularly scheduled inspections.

Attachment 2

Pilgrim Nuclear Power Station
License No. DPR-35 (Docket No. 50-293)

License Renewal Application

License Renewal Commitment

This table identifies actions discussed in this letter that Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are **not** commitments.

ITEM	COMMITMENT	TYPE (Check one) ONE-TIME ACTION	TYPE (Check one) CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (If Required)
51	Enhance the Structures Monitoring Program to invoke quantitative acceptance criteria for inspections of concrete structures in accordance with ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" prior to the period of extended operation (PEO).		X	As stated in the commitment.