Entergy

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Stephen J. Bethay
Director, Nuclear Assessment

October 6, 2006

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 9

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

LETTER NUMBER: 2.06.089

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 six attachments. Attachment A contains the list of revised regulatory commitments. Attachment B contains the response to the requests for additional information (RAIs) on aging management review in LRA Section 3.2 Engineered Safety Features, conveyed in NRC letter dated September 8, 2006. Attachment C contains the response to the RAIs on time limited aging analysis in LRA Section 4.2 Reactor Vessel Neutron Embrittlement, conveyed in NRC letter dated September 8, 2006. Attachment D contains the response to the RAIs on metal fatigue in LRA Section 4.3.1.2 Reactor Vessel Internals, conveyed in NRC letter dated September 7, 2006. Attachment E contains population dose risk reduction for severe accident mitigation alternatives requested in a telephone conference call with the NRC license renewal staff on September 26, 2006. Attachment F contains changes to the LRA and other changes and clarifications stemming from telephone conference calls with the NRC license renewal staff on September 6, 2006 and September 25, 2006, and request for clarification of commitments identified by the NRC license renewal staff on October 4, 2006.

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 October 06, 2006.

Bryan Ford

for Stephen J. Bethay
Director, Nuclear Safety Assessment

DWE/dl
Attachments: (as stated)
cc: see next page

A119

cc: with Attachments

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ATTACHMENT A to Letter 2.06.089
(7 pages)

Revised List of Regulatory Commitments

Revised List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
1	Implement the Buried Piping and Tanks Inspection Program as described in LRA Section B.1.2.	June 8, 2012	Letter 2.06.003	B.1.2/ Audit Item 320
2	Enhance the implementing procedure for ASME Section XI inservice inspection and testing to specify that the guidelines in Generic Letter 88-01 or approved BWRVIP-75 shall be considered in determining sample expansion if indications are found in Generic Letter 88-01 welds.	June 8, 2012	Letter 2.06.003	B.1.6/ Audit Item 320
3	Inspect fifteen (15) percent of the top guide locations using enhanced visual inspection technique, EVT-1, within the first 18 years of the period of extended operation, with at least one-third of the inspections to be completed within the first six (6) years and at least two-thirds within the first 12 years of the period of extended operations. Locations selected for examination will be areas that have exceeded the neutron fluence threshold.	As stated in the commitment	Letters 2.06.064 and 2.06.081	B.1.8/ Audit Items 155, 320
4	Enhance the Diesel Fuel Monitoring Program to include quarterly sampling of the security diesel generator fuel storage tank. Particulates (filterable solids), water and sediment checks will be performed on the samples. Filterable solids acceptance criteria will be = 10 mg/l. Water and sediment acceptance criteria will be = 0.05%.	June 8, 2012	Letters 2.06.003 and 2.06.089	B.1.10/Audit Items 320, 566
5	Enhance the Diesel Fuel Monitoring Program to install instrumentation to monitor for leakage between the two walls of the security diesel generator fuel storage tank to ensure that significant degradation is not occurring.	June 8, 2012	Letter 2.06.057	B.1.10/ Audit Items 155, 320
6	Enhance the Diesel Fuel Monitoring Program to specify acceptance criterion for UT measurements of emergency diesel generator fuel storage tanks (T-126A&B).	June 8, 2012	Letter 2.06.003	B.1.10/ Audit Items 165, 320

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
7	Enhance Fire Protection Program procedures to state that the diesel engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be enhanced to verify that the diesel engine did not exhibit signs of degradation while it was running; such as fuel oil, lube oil, coolant, or exhaust gas leakage. Also, enhance procedures to clarify that the diesel-driven fire pump engine is inspected for evidence of corrosion in the intake air, turbocharger, and jacket water system components as well as lube oil cooler. The jacket water heat exchanger is inspected for evidence of corrosion or buildup to manage loss of material and fouling on the tubes. Also, the engine exhaust piping and silencer are inspected for evidence of internal corrosion or cracking.	June 8, 2012	Letter 2.06.064	B.1.13.1/ Audit Items 320, 378
8	Enhance the Fire Protection Program procedure for Halon system functional testing to state that the Halon 1301 flex hoses shall be replaced if leakage occurs during the system functional test.	June 8, 2012	Letter 2.06.003	B.1.13.1/ Audit Item 320
9	Enhance Fire Water System Program procedures to include inspection of hose reels for corrosion. Acceptance criteria will be enhanced to verify no significant corrosion.	June 8, 2012	Letter 2.06.003	B.1.13.2/ Audit Item 320
10	Enhance the Fire Water System Program to state that a sample of sprinkler heads will be inspected using guidance of NFPA 25 (2002 Edition) Section 5.3.1.1.1. NFPA 25 also contains guidance to repeat this sampling every 10 years after initial field service testing.	June 8, 2012	Letter 2.06.003	B.1.13.2/ Audit Item 320
11	Enhance the Fire Water System Program to state that wall thickness evaluations of fire protection piping will be performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion. These inspections will be performed before the end of the current operating term and at intervals thereafter during the period of extended operation. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function.	June 8, 2012	Letter 2.06.003	B.1.13.2/ Audit Item 320
12	Implement the Heat Exchanger Monitoring Program as described in LRA Section B.1.15.	June 8, 2012	Letter 2.06.003	B.1.15/ Audit Item 320

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
13	Enhance the Instrument Air Quality Program to include a sample point in the standby gas treatment and torus vacuum breaker instrument air subsystem in addition to the instrument air header sample points.	June 8, 2012	Letter 2.06.003	B.1.17/ Audit Item 320
14	Implement the Metal-Enclosed Bus Inspection Program as described in LRA Section B.1.18.	June 8, 2012	Letter 2.06.003	B.1.18/ Audit Item 320
15	Implement the Non-EQ Inaccessible Medium-Voltage Cable Program as described in LRA Section B.1.19. Include developing a formal procedure to inspect manholes for in-scope medium voltage cable.	June 8, 2012	Letter 2.06.003	B.1.19/ Audit items 311, 320
16	Implement the Non-EQ Instrumentation Circuits Test Review Program as described in LRA Section B.1.20.	June 8, 2012	Letter 2.06.003	B.1.20/ Audit Item 320
17	Implement the Non-EQ Insulated Cables and Connections Program as described in LRA Section B.1.21.	June 8, 2012	Letter 2.06.003	B.1.21/ Audit Item 320
18	Enhance the Oil Analysis Program to periodically change CRD pump lubricating oil. A particle count and check for water will be performed on the drained oil to detect evidence of abnormal wear rates, contamination by moisture, or excessive corrosion.	June 8, 2012	Letter 2.06.003	B.1.22/Audit Item 320
19	Enhance Oil Analysis Program procedures for security diesel and reactor water cleanup pump oil changes to obtain oil samples from the drained oil. Procedures for lubricating oil analysis will be enhanced to specify that a particle count and check for water are performed on oil samples from the fire water pump diesel, security diesel, and reactor water cleanup pumps.	June 8, 2012	Letter 2.06.003	B.1.22/ Audit Item 320
20	Implement the One-Time Inspection Program as described in LRA Section B.1.23. This includes destructive or non-destructive examination of one (1) socket welded connection using techniques proven by past industry experience to be effective for the identification of cracking in small bore socket welds. Should an inspection opportunity not occur (e.g., socket weld failure or socket weld replacement), a susceptible small-bore socket weld will be examined either destructively or non-destructively prior to entering the period of extended operation.	June 8, 2012	Letter 2.06.003	B.1.23/ Audit Items 219, 320
21	Enhance the Periodic Surveillance and Preventive Maintenance Program as necessary to assure that the effects of aging will be managed as described in LRA Section B.1.24.	June 8, 2012	Letter 2.06.003	B.1.24/ Audit Item 320

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
22	Enhance the Reactor Vessel Surveillance Program to proceduralize the data analysis, acceptance criteria, and corrective actions described in LRA Section B.1.26.	June 8, 2012	Letter 2.06.003	B.1.26/ Audit Item 320
23	Implement the Selective Leaching Program in accordance with the program as described in LRA Section B.1.27.	June 8, 2012	Letter 2.06.003	B.1.27/ Audit Item 320
24	Enhance the Service Water Integrity Program procedure to clarify that heat transfer test results are trended.	June 8, 2012	Letter 2.06.003	B.1.28/ Audit Item 320
25	Enhance the Structures Monitoring Program procedure to clarify that the discharge structure, security diesel generator building, trenches, valve pits, manholes, duct banks, underground fuel oil tank foundations, manway seals and gaskets, hatch seals and gaskets, underwater concrete in the intake structure, and crane rails and girders are included in the program. In addition, the Structures Monitoring Program will be revised to require opportunistic inspections of inaccessible concrete areas when they become accessible.	June 8, 2012	Letter 2.06.003	B.1.29.2/ Audit Items 238, 320
26	Enhance Structures Monitoring Program guidance for performing structural examinations of elastomers (seals, gaskets, seismic joint filler, and roof elastomers) to identify cracking and change in material properties.	June 8, 2012	Letter 2.06.003	B.1.29.2/ Audit Item 320
27	Enhance the Water Control Structures Monitoring Program scope to include the east breakwater, jetties, and onshore revetments in addition to the main breakwater.	June 8, 2012	Letter 2.06.003	B.1.29.3/ Audit Item 320
28	Enhance System Walkdown Program guidance documents to perform periodic system engineer inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).	June 8, 2012	Letter 2.06.057	B.1.30/ Audit Items 320, 327
29	Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Section B.1.31.	June 8, 2012	Letter 2.06.003	B.1.31/ Audit Items 257, 320

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
30	Perform a code repair of the CRD return line nozzle to cap weld if the installed weld repair is not approved via accepted code cases, revised codes, or an approved relief request for subsequent inspection intervals.	June 30, 2015	Letter 2.06.057	B.1.3/ Audit Items 141, 320
31	At least 2 years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for BWRs of the PNPS vintage, PNPS will implement one or more of the following: (1) Refine the fatigue analyses to determine valid CUFs less than 1 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations, including NUREG/CR-6260 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 external loads may be used if demonstrated applicable to PNPS. 4. An analysis using an NRC-approved version of the ASME code of NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. (2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC). (3) Repair or replace the affected locations before exceeding a CUF of 1.0. Should PNPS select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.	June 8, 2012 June 8, 2010 for submitting the aging management program if PNPS selects the option of managing the affects of aging due to environmentally assisted fatigue.	Letters 2.06.064 and 2.06.081	4.3.3/ Audit Items 302, 346
32	Implement the enhanced Bolting Integrity Program described in Attachment C of Pilgrim License Renewal Application Amendment 5 (dated July 19, 2006, 2.06.064).	June 8, 2012	Letters 2.06.064 and 2.06.081	Audit items 364, 373, 389, 390, 432, 443, 470

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
33	PNPS will inspect the inaccessible jet pump thermal sleeve and core spray thermal sleeve welds if and when the necessary technique and equipment become available and the technique is demonstrated by the vendor, including delivery system.	As stated in the commitment	Letter 2.06.057	Audit Items 320, 488
34	Within the first 6 years of the period of extended operation and every 12 years thereafter, PNPS will inspect the access hole covers with UT methods. Alternatively, PNPS will inspect the access hole covers in accordance with BWRVIP guidelines should such guidance become available.	June 8, 2018	Letter 2.06.057 and 2.06.089	Audit Items 320, 461
35	At least 2 years prior to entering the period of extended operation, for reactor vessel components, including the feedwater nozzles, PNPS will implement one or more of the following: (1) Refine the fatigue analyses to determine valid CUFs less than 1. Determine valid CUFs based on numbers of transient cycles projected to be valid for the period of extended operation. Determine CUFs in accordance with an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). (2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC). (3) Repair of replace the affected locations before exceeding a CUF of 1.0. Should PNPS select the option to manage the aging effects due to fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.	June 8, 2012 June 8, 2010 for submitting the aging management program if PNPS selects the option of managing the affects of aging.	Letters 2.06.064 and 2.06.081	Audit Item 345
36	To ensure that significant degradation on the bottom of the condensate storage tank is not occurring, a one-time ultrasonic thickness examination in accessible areas of the bottom of the condensate storage tank will be performed. Standard examination and sampling techniques will be utilized.	June 8, 2012	Letter 2.06.057	Audit Items 320, 363

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
37	The BWR Vessel Internals Program includes inspections of the steam dryer. Inspections of the steam dryer will follow the guidelines of BWRVIP-139 and General Electric SIL 644 Rev. 1.	June 8, 2012	Letter 2.06.089	A.2.1.8/ Conference call on September 25, 2006
38	Enhance the Diesel Fuel Monitoring Program to include periodic ultrasonic thickness measurement of the bottom surface of the diesel fire pump day tank. The first ultrasonic inspection of the bottom surface of the diesel fire pump day tank will occur prior to the period of extended operation, following engineering analysis to determine acceptance criteria and test locations. Subsequent test intervals will be determined based on the first inspection results.	June 8, 2012	Letter 2.06.089	B.1.10/ Audit Item 565

ATTACHMENT B to Letter 2.06.089
(2 pages)

Response to Requests for Additional Information on Aging Management Review in LRA Section 3.2 Engineered Safety Features

3.2 Engineered Safety Features Systems

RAI 3.2.1

Piping of the Reactor Core Isolation Cooling System is subject to flow accelerated corrosion.

The applicant in LRA Table 3.2.1-19 - engineered safety feature (ESF) and Table 3.2.1 credit the plant-specific Periodic Surveillance and Preventive Maintenance Program for the management of loss of materials (wall thinning). The GALL report recommends using the Flow-Accelerated Corrosion Program, XI.M17, to manage wall thinning. The Periodic Surveillance and Preventive Maintenance Program provides for inspection for material loss (wall thinning) every 5 years but no Monitoring and Trending activities to predict areas of high wall thinning rates or for trending of thinning as does XI.M17. Please provide justification for not providing for monitoring and trending of wall thinning for reactor core isolation cooling (RCIC) piping.

RAI 3.2.1 Response

Flow-accelerated corrosion is not an aging effect requiring management for RCIC system components in LRA Table 3.2.2-5, "Reactor Core Isolation Cooling System (RCIC) Summary of Aging Management Evaluation," due to infrequent system operation. As stated in LRA Section B.1.14, the Flow-Accelerated Corrosion program applies to safety-related and nonsafety-related carbon steel components in systems containing high-energy fluids carrying two-phase or single-phase high-energy fluid ≥ 2% of plant operating time.

Portions of RCIC steam supply and exhaust piping downstream of the strainers and steam traps are subjected to constricted flow and are therefore susceptible to erosion. The piping line item in LRA Table 3.2.2-5 that references Table 1 Item 3.2.1-19 represents loss of material due to erosion, not flow-accelerated corrosion, for these sections of piping. As indicated in the table, the plant-specific Periodic Surveillance and Preventive Maintenance Program manages this loss of material due to erosion. Line item 3.2.1-19 in Table 3.2.1, "Summary of Aging Management Programs for Engineered Safety Features Evaluated in Chapter V of NUREG-1801," indicates that the plant-specific Periodic Surveillance and Preventive Maintenance Program includes periodic non-destructive evaluations to identify wall thinning, thereby managing loss of material due to erosion for this piping.

As described in LRA Section B.1.24, under the Monitoring and Trending attribute, preventive maintenance and surveillance testing activities provide for monitoring and trending of aging degradation. Inspection and testing intervals are established such that they provide for timely detection of component degradation. Inspection and testing intervals are dependent on component material and environment and take into consideration industry and plant-specific operating experience and manufacturers' recommendations. Therefore, monitoring and trending of applicable aging mechanisms for RCIC piping is provided by the Periodic Surveillance and Preventive Maintenance Program.

RAI 3.2.2

The applicant in Table 3.2.1 Item 18 of the PNPS LRA stated that none of the ESF system components are within the scope of the BWR Stress Corrosion Cracking Program. Please provide the details that justify this statement.

RAI 3.2.2 Response

As stated in LRA Section B.1.6, the BWR Stress Corrosion Cracking Program applies to reactor coolant pressure boundary components made of stainless steel or CASS (cast austenitic stainless steel). As described in LRA Section 2.3.1.3, ESF system components that are part of the reactor coolant pressure boundary (Class I) are included in the aging management review of the reactor coolant pressure boundary provided in Table 3.1.2-3.

ATTACHMENT C to Letter 2.06.089
(4 pages)

Response to Requests for Additional Information on Time Limiting Aging Analysis in LRA Section 4.2 Reactor Vessel Neutron Embrittlement

RAI 4.2.2-1

Section 4.2.2 of the Pilgrim Nuclear Power Station (PNPS) License Renewal Application (LRA), "Pressure-Temperature [P-T] Limits," states that in a license amendment request dated December 4, 2002, PNPS requested to use the present P-T limit curves through the end of operating cycle 16, which corresponds to approximately 23 effective full power years (EFPY) of facility operation. The end of operating cycle 16 is expected to occur in 2007. Section 4.2.2 also states that, in this December 4, 2002 submittal, PNPS committed to develop and submit updated P-T limit curves and revised fluence calculations based on an NRC-approved calculation method that adheres to Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," prior to the end of operating cycle 16. License Amendment 197 granted this request in 2003. Section 4.2.2 then states that recent fluence calculations that were done per RG 1.190 confirm that the fluence for 54 EFPY is less than the fluence used to calculate the P-T limits that were approved for use only through the end of operating cycle 14. Based on the above statements, you conclude that the current Technical Specification (TS) P-T limit curves remain valid for the period of extended operation.

(a) Please confirm whether the P-T limit curves currently established in the PNPS TSs expire at the end of operating cycle 16 (23 EFPY).

(b) Please explain why the recent fluence calculations that were done per RG 1.190 confirm that the fluence for 54 EFPY is less than the fluence used to calculate the P-T limits that were approved for use through the end of cycle 14. Explain how this information was used to determine that the existing P-T limits remain valid for the period of extended operation.

(c) The staff does not require the P-T limit curves for the extended period of operation to be submitted as part of the applicant's LRA for this time-limited aging analysis (TLAA). However, the staff does require NRC approval of the P-T limit curves for the extended period of operation prior to the expiration of the P-T limit curves for 32 EFPY. Please state when you intend to submit P-T limit curves for NRC approval for the extended licensed period of operation (54 EFPY).

RAI 4.2.2-1 RESPONSE

(a) Yes, the P-T limit curves in the PNPS Technical Specifications (Figures 3.6-1, 3.6-2 and 3.6-3) are approved through the end of operating cycle 16.

(b) The fluence value on which the (through cycle 16) P-T curves were based was 9.95×10^{17} n/cm^2, (e >1 MeV). [Entergy letter, R. Bellamy to NRC, "Request for Technical Specification Change Concerning Pressure-Temperature Limit Curves of Figure 3.6.1, 3.6.2, and 3.6.3," dated November 22, 2000, 2.00.080.] The projected fluence value for 54 EFPY, from Section 4.2.1 of the LRA, is 8.4×10^{17} n/cm^2, (e >1 MeV). The original fluence was not calculated in accordance with Regulatory Guide 1.190, and in fact, was very conservative. The new fluence value for 54 EFPY is calculated in accordance with Regulatory Guide 1.190. This information was used to determine that the existing P-T limits remain valid for the period of extended operation in accordance with NUREG-1800, Section 4.2.2.1.3.1, which states, "The existing P-T limits are valid during the period of extended operation because the neutron fluence projected to the end of the period of extended operation is bound by the fluence assumed in the existing analysis."

(c) PNPS has submitted proposed P-T curves for approval prior to the expiration of the approved P-T curves in the Technical Specifications (cycle 16). These curves are requested for approval to 34 EFPY of facility operation. New curves will be submitted for approval prior to expiration of the proposed curves.

RAI 4.2.4-1

Table 4.2-2 of the PNPS LRA lists initial RT_{NDT} values for the PNPS RV beltline materials. The initial RT_{NDT} values for Lower Intermediate Shell Plate G-3108-1 (Heat No. C-2921-2) and Lower Intermediate Shell Plate G-3108-3 (Heat No. C-2945-2) are less conservative than the corresponding initial RT_{NDT} values established in the NRC staff's reactor vessel integrity database (RVID) for these materials. Section 4.2.4 of the PNPS LRA states that, "initial RT_{NDT} values are from report SIR-00-82, which was submitted in 2001 as part of the PNPS P-T limit change request (Reference 4.2-5)." Reference 4.2-5 points to the April 13, 2001, license amendment issued by the NRC authorizing revised P-T limit curves. Please provide additional information that points to where the NRC staff authorized the use of the specific initial RT_{NDT} values listed in Table 4.2-4 for determining the adjusted reference temperature (ART) values for the PNPS reactor vessel (RV) beltline materials.

RAI 4.2.4-1 RESPONSE

As stated in the LRA, PNPS provided revised values of RT_{NDT} in response to an RAI on the request supporting a proposed change that resulted in the current P-T limit technical specification. [Entergy letter, R. Bellamy to NRC, "Additional Information Related to Pilgrim Technical Specification Change Concerning Pressure-Temperature Limit Curves of Figures 3.6.1.2 and 3," dated January 30, 2001, 2.01.014.] The technical detail was in the attachment to this letter, Structural Integrity Associates report SIR-00-082, "Updated Evaluation of Reactor Pressure Vessel Materials Properties for Pilgrim Nuclear Power Station." The SER accepting the new curves did not specifically state that the new RT_{NDT} could be used in place of the values then in RVID2; however, the NRC approved the technical specification change. [NRC letter, A. Wang to R. Bellamy, "Pilgrim Nuclear Power Station – Issuance of Amendment RE: Pressure-Temperature Limit Curves (TAC No. MB0561)," dated April 13, 2001.]

RAI 4.2.4-2

The %Cu and chemistry factor (CF) values for Lower Shell Axial Welds 2-338A, B, and C from LRA Table 4.2-2 are less conservative than the corresponding %Cu and CF values that were established in the staff's RVID for these welds. Please provide the following information:

(a) verification of whether the %Cu and CF values listed in Table 4.2-1 are valid for the above welds,

(b) justification for the use of these chemistry data for the above welds, including the source of the data, and a specific reference for the documentation/analysis demonstrating that these chemistry data represent the best available estimate of the weld chemistries.

RAI 4.2.4-2 RESPONSE

(a) The values come from SIR-00-082 discussed in response to RAI 4.2.4-1.

(b) SIR-00-082 discussed in response to RAI 4.2.4-1 provided the requested information.

RAI 4.2.4-3

Lower Intermediate / Upper Shell Circumferential Weld 3-339B (Heat No. 13253) is listed in the NRC staff's RVID. However this weld is not represented in LRA Table 4.2-2 or (LRA Table 4.2-1). Please resolve this discrepancy.

RAI 4.2.4-3 RESPONSE

RVID2 used essentially the same (peak) fluence for all locations. In the LRA, fluence for individual locations was used. The lower intermediate / upper shell Circumferential Weld 3-339B had a fluence of only 5×10^{15} n/cm^2 (e>1 Mev) and was therefore deleted from the list.

RAI 4.2.5-1

Section 4.2.5 of the PNPS LRA addresses the TLAA for the RV Circumferential Weld Examination Relief. Table 4.2-3 of the LRA compares the limiting RV circumferential weld parameters for PNPS to those used in the NRC evaluation of the BWRVIP-05 report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations." The PNPS limiting RV circumferential weld parameters are based on Lower Intermediate / Lower Shell Circumferential Weld 1-334 (Heat No. 21935), which is the only circumferential weld represented in LRA Table 4.2-2. However, as discussed in RAI 4.2.4-3, the NRC staff's RVID also lists Lower Intermediate / Upper Shell Circumferential Weld 3-339B (Heat No. 13253) as one of the RV welds for PNPS. Furthermore, the chemistry and CF data for this weld are more limiting than for the Circumferential Weld 1-334. Please explain why this TLAA did not address Lower Intermediate / Upper Shell Circumferential Weld 3-339B (Heat No. 13253).

RAI 4.2.5-1 RESPONSE

RVID2 used essentially the same (peak) fluence for all locations. In the LRA, fluence for individual locations was used. The Lower Intermediate / Upper Shell Circumferential Weld 3-339B had a fluence of only 5×10^{15} n/cm^2 (e>1 Mev) and was therefore deleted from the list.

RAI 4.2.5-2

The NRC staff requires that a request for relief from the American Society of Mechanical Engineer Boiler and Pressure Vessel Code (ASME Code) RV circumferential shell weld examination requirements be submitted prior to the beginning of the extended period of operation. Please state whether you intend to apply for relief from the ASME Code RV circumferential weld examination requirements for the extended licensed period of operation. State when you plan to submit this relief request.

RAI 4.2.5-2 RESPONSE

In accordance with 10 CFR 50.55a(g)5(iv), PNPS will submit this request for each ASME Section XI Inservice Inspection ten-year interval within 12 months after the completion of the prior interval.

RAI 4.2.5-3

In the July 28, 1998 SER on BWRVIP-05, the NRC staff concluded that examination of the RV circumferential shell welds would need to be performed if the corresponding volumetric examinations of the RV axial shell welds revealed the presence of an age-related degradation mechanism. Confirm whether or not previous volumetric examinations of the RV axial shell welds have shown any indication of cracking or other age-related degradation mechanisms in the welds.

RAI 4.2.5-3 RESPONSE

Examinations of the PNPS reactor vessel axial shell welds have not identified any indications or other age-related degradation mechanisms in the welds.

RAI 4.2.6-1

The limiting axial weld failure probability calculated by the NRC staff in the BWRVIP-05 SER is based on the assumption that "essentially 100 percent" (i.e., greater than 90 percent) examination coverage of all RV axial welds can be achieved in accordance with ASME Code, Section XI requirements.

State whether your ISI examinations achieve "essentially 100 percent" (i.e., greater than 90 percent) overall examination coverage for the RV axial welds for the duration of the current licensed operating period. If less than 90 percent overall examination coverage is achieved for the RV axial welds, revise this TLAA to account for the effects of the limited scope examination coverage.

RAI 4.2.6-1 RESPONSE

Due to various obstructions within the reactor vessel, Pilgrim (PNPS) has not been able to inspect "essentially 100%" of the reactor vessel beltline axial welds. PNPS identified the exact coverage of each weld and the inspection interferences, and requested relief from the requirement to inspect at least 90% of the welds[1]. Evaluation of the data in this letter shows that approximately 83% of the axial weld length in the beltline region was inspected (67% of the total welds were inspected but portions of the un-inspected welds are not in the beltline). Although less than 90%, the actual coverage should identify any pattern of degradation, particularly in the beltline region, where approximately 83% of the axial weld length is located.

The NRC granted the relief request for less than 90% coverage[2]. In part, the enclosed safety evaluation, in Section 6.3 Limitations to Examination, states "... a significant portion of the weld volume was examined without any evidence of unacceptable degradation. The absence of unacceptable indications in any of the examined welds supports the licensee's contention that additional examinations to achieve at least 90% coverage of every weld would present a hardship and produce unnecessary radiological exposure to personnel."

The effect of this reduced inspection (83% vs. 90%) on the axial weld failure probability would be small, but PNPS has not attempted to quantify that effect. Table 4.2-4 of the LRA shows a large margin between the 54.9 °F mean adjusted reference temperature for PNPS versus the 172.4 °F / 91.0 °F mean adjusted reference temperatures for the CEOG and Clinton plants used in the NRC SER for BWRVIP-05 to determine axial weld failure probability. Inspecting 83%, instead of 90%, of the welds is unlikely to offset this large margin.

Therefore, a revision to this TLAA is not necessary.

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1 BECo letter, E.T. Boulette to U. S. NRC, "Results of Augmented Examination of the RPV Shell Welds and Relief Request Pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5)," dated September 20, 1995, 2.95.099

2 NRC letter, S.F. Shankman to E.T. Boulette, "Pilgrim Nuclear Power Station - Request for Authorization of Alternative Reactor Pressure Vessel Examinations (TAC No. M93724)," dated March 26, 1996

ATTACHMENT D to Letter 2.06.089
(4 pages)

Response to Requests for Additional Information on Metal Fatigue in LRA Section 4.3.1.2 Reactor Vessel Internals

4.3.1.2 Reactor Vessel Internals

RAI 4.3.1.2-1

Control rod drive (CRD) return line nozzle-to-end cap weld: Regarding the CRD return line nozzle-to-end cap weld repair, your Project Report LRPD-06, "Pilgrim NPS License Renewal Project – Time-Limited Aging Analyses, Mechanical Fatigue," Response 2.4 refers to Relief Request PRR-36 and concludes, "This relief did not involve any analyses based on time-limited assumptions and therefore is not a TLAA." PRR-36 was submitted by letters dated October 1, 3, and 8, 2004 and July 12, 2004, for relief from certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements pertaining to the repair of the nozzle-to-end cap weld with a detected flaw and the associated nondestructive examinations. Alternatively, PRR-36 proposed to use ASME Code Case N-504-2, "Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping," with modifications to perform the repair. The request was approved in a safety evaluation (SE) dated February 25, 2005. ASME Code Case N-504-2 (g)(2) requires a flaw evaluation be performed on the repaired component such that "The evaluation should demonstrate that the requirements of IWB-3640...are satisfied for the design life of the repair, considering potential flaw growth due to fatigue and the mechanism believed to have caused the flaw. The flaw growth evaluation shall be performed in accordance with Appendix C." Explain how Entergy meet the ASME Code Case N-504-2 requirement on performing a flaw evaluation that considers fatigue and the mechanism believed to have caused the flaw. If applicable you may provide a document showing that the weld overlay region adjacent to the interface is in the compressive stress zone.

RAI 4.3.1.2-1 Response

The cause of the failure was interdendritic stress corrosion cracking (IDSCC) in the weld of the cap to the nozzle. Crack initiation was due to a through wall repair and internal diameter grinding that was performed to remove weld fabrication flaws during installation of the nozzle cap.

The repair of the crack found in 2003 was a full structural weld overlay. This type of repair has been shown by industry studies and experience over the past 25 years to produce compressive weld residual stresses in the inner portion of the component wall. This weld residual stress distribution will inhibit the initiation of new flaws due to stress corrosion cracking (SCC), and also will inhibit the growth of any existing flaws.

The overlay material is alloy 52 (UNS 6052), which has a chrome content of ~ 30% and is highly resistant to SCC.

The design basis for the repair was the assumption that the observed flaw was entirely through the original component wall. The actual flaw was leaking in one location. The maximum flaw length was about 4" at the internal diameter.

The weld overlay material, as noted above, is resistant to SCC-driven flaw propagation due to its composition and grain structure. Therefore, further propagation due to the initiating mechanism is not expected, and the flaw growth evaluation was not based on a time-limited assumption.

Reinspection of the repair will be per BWRVIP-75 Category E requirements which are to examine by UT 25% of Category E overlays every 10 years, and also to perform a UT exam within three refueling outages of the October 2003 repair (i.e. scheduled for RFO17 in 2009).

These inspections will verify the absence of any flaw growth that would invalidate the overlay design basis.

Fatigue crack growth is also negligible because the only cycling is steam cycling, and there are very few such cycles over the life of the plant. Further, there are no piping loads since the nozzle has been capped.

RAI 4.3.1.2-2

Reactor recirculation nozzle thermal sleeves regarding the flaws on reactor recirculation nozzle thermal sleeves, LRPD-06 Response (sic) 2.4 refers to a flaw growth analysis in NEDC-30730 and concludes, "The NRC reviewed and accepted the analysis as documented in an SER (Ref. 4.2.21). As this analysis is only based on 18 months, it is not a TLAA." The cited SE was issued on December 4, 1984. As you stated, the crack growth analysis is for 18 months. One of the six criteria specified in 10 CFR 54.3(a) for classifying an analysis as a TLAA is the analysis "[i]nvolve time-limited assumptions defined by the current operating term, for example, 40 years." The meaning of a crack growth analysis based on 18 months is that the structural integrity of reactor recirculation nozzle thermal sleeves is not only a concern for the extended period of operation but also a concern for the remaining period of operation under the current 40-year license. Therefore, Entergy needs to consider this as a TLAA and address the following:

For the License Renewal Application:

(1) Confirm whether Report PMA86-07, "Pilgrim Nuclear Power Station Recirculation Inlet Thermal Sleeve Mock-up Fabrication and Evaluation," dated October 1986 had been reviewed by NRC staff.

(2) Identify the SE which accepts use of hydrogen water chemistry as the mitigating method and as the basis for Entergy to operate with the flaws on the thermal sleeves beyond 1987.

(3) Provide an analysis of the inspection results on these thermal sleeves obtained from 1987 to date.

(4) Provide the end-of-extended-period-of-operation (60 years) flaw length of the circumferential through-wall flaw which was 32 percent circumference in 1987 (per the December 4, 1984, SE for the worst flaw among the detected recirculation nozzle thermal sleeve cracks) and perform a stability analysis for this flaw.

(5) If the stability analysis of effort (4) shows that the predicted end-of-extended-period-of-operation through-wall flaw length does not meet the ASME Code Section XI margin, provide an impact evaluation on operation and structural integrity of other components due to a broken thermal sleeve piece of a reasonable size.

(6) Provide an inspection plan for these detected thermal sleeve flaws in the extended period of operation.

For current operation till the end of the 40-year operation:

(7) Discuss the adequacy of the inspection plan for recirculation nozzle thermal sleeves for the remaining period of 40-year operation.

(8) Provide the end of the 40-year operation flaw length of the circumferential through-wall flaw which was 32 percent circumference in 1987 and perform a stability analysis for this flaw.

RAI 4.3.1.2-2 Response

The original evaluation in NEDC-30730 was for 18 months of operation. Subsequently, an additional analysis, General Electric Report SASR 87-05, Rev. 1, performed in January 1987 extended the acceptable operating time with normal water chemistry (NWC) to 35,000 hours. This time was increased to 65,000 NWC hours by a revised evaluation, GENE-523-A143-1295, Rev. 1, June 1996. The PNPS operating time with NWC is currently less than 65,000 hours. The majority of operating hours are with hydrogen water chemistry (HWC) operation.

Because the thermal sleeve is not part of the reactor coolant pressure boundary, its failure would not compromise the pressure boundary.

The following are responses to items (1) through (8).

(1) The subject report, General Electric (GE) Report PMA86-07 *Pilgrim Nuclear Power Station Recirculation Inlet Thermal Sleeve Mockup Fabrication and Evaluation,* dated October 1986, was submitted to the NRC by letter dated January 2, 1987 (2.87.003) as part of correspondence on NRC Order dated August 26, 1983 (i.e. IGSCC Order). The letter (dated January 2, 1987) is referenced in NRC safety evaluation report (SER) dated January 29, 1991 (TAC NO. 60939) that was issued as part of License Amendment 134, to Pilgrim Station. In addition to referencing the letter, the Mechanical Equipment section of the SER indicates the NRC was provided with the results of "this thermal sleeve study that indicated IGSCC as the most likely cracking mechanism and that hydrogen water chemistry was planned as the method to mitigate IGSCC at Pilgrim." Although the report is not specifically identified (i.e. by number or title or date) in the SER, it appears the report was or had been reviewed by the staff of the NRC Office of Nuclear Reactor Regulation when the SER for License Amendment 134 was prepared. Moreover, the letter (January 2, 1987) is referenced and the report is specifically identified by title and date in NRC Inspection Report 50-293/87-46, dated December 7, 1987, on pages 12 – 13. In part, the inspection report states: "...The NRC inspector reviewed the General Electric (GE) report ...." Although the report number, PMA86-07, is not specifically identified in the inspection report, it is evident the report was reviewed by the NRC inspector.

(2) There is no safety evaluation which accepts use of HWC as the mitigating method and as the basis for Pilgrim to operate with the flaws on the thermal sleeves beyond 1987. No safety evaluation is required since failure of the thermal sleeve has no bearing on pressure boundary integrity. A submittal was made to the NRC by letter dated August 4, 1988 (2.88.119) that provided response to Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." In the Generic Letter all BWR licensees were requested to address the staff positions for conformance or propose alternative measures, and supply information relating to piping replacement, inspection, repair and leakage due to IGSCC in BWR reactor coolant pressure boundary piping. The response letter described Pilgrim's conformance to GL 88-01, including information relating to scope of replacement, repair, inspection and implementation of alternate measures. Discussion was made in that submittal that HWC was being implemented in refueling outage 7 (1986 - 1988). NRC letter dated April 26, 1990 issued a safety evaluation report (SER) that addressed the response to GL 88-01. The SER (TAC No. 69153) acknowledged that HWC has been implemented at PNPS and that HWC was expected to provide protection for the reactor coolant pressure boundary. Although the thermal sleeve is not part of the reactor coolant pressure boundary, it is exposed to reactor recirculation flow and is protected by HWC from IGSCC. The failure of the thermal sleeve due to cracking would not compromise the pressure boundary.

(3) There has been no subsequent inspection of the recirculation inlet thermal sleeves because there is no practical method to perform an inspection.

(4) Report GENE-523-A143-1295, Rev. 1, June 1996, determined that there could be 65,000 hours of operation with normal water chemistry before limits for the thermal sleeves are reached. The analysis is based on the conservative assumption of four through wall cracks 2.94 inches long, the remaining circumference has 0.17 deep cracks everywhere and a limit load safety factor of 2.77. Actual crack depth was never determined since the cracks were found by a penetrant testing examination.

(5) The thermal sleeve is not pressure boundary so its failure would not compromise the pressure boundary. Failure of the thermal sleeve would be detected as a change in differential pressure of the affected jet pumps. There would be some slight movement but the thermal sleeve would remain within the nozzle. The movement of the riser pipe is restricted by the shroud. In addition, the cracks are at the outer end of the outer thermal sleeve. A full circumferential failure would not allow inward movement because the inner end of the outer thermal sleeve is welded to the nozzle and this would restrain movement.

(6) Inspections will be performed per BWRVIP guidelines subject to availability of inspection techniques and equipment.

(7) An inspection plan will be implemented per BWRVIP guidelines when equipment to perform the inspection becomes available to the industry. A complete circumferential failure and movement of the thermal sleeve would be detected by a change in differential pressure of the affected jet pumps. As discussed previously, the thermal sleeve would stay within the nozzle allowing for orderly shutdown and repair.

(8) Refer to the discussion in item (4) above.

ATTACHMENT E to Letter 2.06.089
(4 pages)

Population Dose Risk Reduction for Severe Accident Mitigation Alternatives
Requested in Telephone Conference with NRC License Renewal Staff on September 26, 2006

The table beginning on the next page contains population dose risk (PDR) reduction in units of % for each Severe Accident Mitigation Alternative (SAMA) and for RAIs 5e, 5f, 5g, and 5h.

To three significant digits, the values for CDF, PDR, and OECR SAMAs 6 (equivalent to 18 and 20), 48, and 52 are as follows:

	Initial Analysis			Re-analysis		
SAMA	CDF	PDR	OECR	CDF	PDR	OECR
6,18,20	6.41E-06	1.35E+01	4.59E+04	6.41E-06	1.46E+01	5.26E+04
48	6.41E-06	1.35E+01	4.59E+04	6.41E-06	1.46E+01	5.26E+04
52	6.40E-06	1.35E+01	4.59E+04	6.40E-06	1.46E+01	5.26E+04
Base	6.41E-06	1.36E+01	4.59E+04	6.41E-06	1.46E+01	5.26E+04

Small benefits could result from minor differences in CDF, PDR, or OECR. For example, slight difference in PDR for SAMA 6 and Base results in a benefit of $2,153 and an upper bound benefit of $12,915 with a multiplier of 6 in the initial analysis supporting Appendix E Attachment E, submitted as part of the License Renewal Application (January 25, 2006). However, there is no such difference for the reanalysis. Therefore, the estimated benefit for SAMA 6 is $0 (zero dollars).

Also, the Reduction in Off-site Economic Cost Risk (OECR) reduction for SAMA 27 on Table RAI.6-1 should be 15.02% (same as RAI 5e) rather than 1.71%.

Reduction in Population Dose Risk (PDR)

SAMA ID	SAMA Description	PDR Reduction (%)
1	Install an independent method of suppression pool cooling.	4.79%
2	Install a filtered containment vent to provide fission product scrubbing.	18.49%
3	Install a containment vent large enough to remove ATWS decay heat.	1.37%
4	Create a large concrete crucible with heat removal potential under the base mat to contain molten core debris.	48.97%
5	Create a water-cooled rubble bed on the pedestal.	48.97%
6	Provide modification for flooding the drywell head.	0.00%
7	Enhance fire protection system and standby gas treatment system hardware and procedures.	1.37%
8	Create a core melt source reduction system.	48.97%
9	Install a passive containment spray system.	4.79%
10	Strengthen primary and secondary containment.	26.03%
11	Increase the depth of the concrete basemat or use an alternative concrete material to ensure melt-through does not occur	0.68%
12	Provide a reactor vessel exterior cooling system	0.00%
13	Construct a building to be connected to primary/ secondary containment that is maintained at a vacuum	1.37%
14	Dedicated Suppression Pool Cooling	4.79%
15	Create a larger volume in containment.	26.03%
16	Increase containment pressure capability (sufficient pressure to withstand severe accidents).	26.03%
17	Install improved vacuum breakers (redundant valves in each line).	0.00%
18	Increase the temperature margin for seals.	0.00%
19	Install a filtered vent	18.49%
20	Provide a method of drywell head flooding.	0.00%
21	Use alternate method of reactor building spray.	1.37%
22	Provide a means of flooding the rubble bed.	22.60%
23	Install a reactor cavity flooding system.	48.97%
24	Add ribbing to the containment shell.	26.03%
25	Provide additional DC battery capacity.	2.74%

Reduction in Population Dose Risk (PDR)

SAMA ID	SAMA Description	PDR Reduction (%)
26	Use fuel cells instead of lead-acid batteries.	2.74%
27	Modification for Improving DC Bus Reliability	16.44%
28	Provide 16-hour SBO injection.	2.74%
29	Provide an alternate pump power source.	5.48%
30	AC Bus Cross-Ties	8.22%
31	Add a dedicated DC power supply.	16.44%
32	Install additional batteries or divisions.	16.44%
33	Install fuel cells.	2.74%
34	DC Cross-Ties	2.05%
35	Extended SBO provisions.	2.74%
36	Locate RHR inside containment.	0.00%
37	Increase frequency of valve leak testing.	0.68%
38	Improve MSIV design.	0.00%
39	Install an independent diesel for the CST makeup pumps.	0.00%
40	Provide an additional high pressure injection pump with independent diesel.	2.05%
41	Install independent AC high pressure injection system.	2.05%
42	Install a passive high pressure system.	2.05%
43	Improved high pressure systems	1.37%
44	Install an additional active high pressure system.	2.05%
45	Add a diverse injection system.	2.05%
46	Increase SRV reseat reliability.	0.68%
47	Install an ATWS sized vent.	1.37%
48	Diversify explosive valve operation.	0.00%
49	Increase the reliability of SRVs by adding signals to open them automatically.	0.68%
50	Improve SRV design.	3.42%
51	Provide self-cooled ECCS pump seals.	0.68%

Reduction in Population Dose Risk (PDR)

SAMA ID	SAMA Description	PDR Reduction (%)
52	Provide digital large break LOCA protection.	0.00%
53	Control containment venting within a narrow band of pressure	4.79%
54	Install a bypass switch to bypass the low reactor pressure interlocks of LPCI or core spray injection valves.	0.68%
55	Improve SSW System and RBCCW pump recovery.	6.85%
56	Provide redundant DC power supplies to DTV valves.	3.42%
57	Proceduralize the use of diesel fire pump hydroturbine in the event of EDG A failure or unavailability.	3.42%
58	Proceduralize the operator action to feed B1 loads via B3 when A5 is unavailable post-trip.	3.42%
59	Provide redundant path from fire protection pump discharge to LPCI loops A and B cross-tie.	17.12%
RAI 5e	Equivalent to SAMA 27	16.44%
RAI 5f	Firewater injection	4.11%
RAI 5g	Reduntant diesel firewater pump	8.22%
RAI 5h	Passive direct torus vent	14.38%

ATTACHMENT F to Letter 2.06.089
(5 pages)

Changes to the LRA Including Changes and Clarifications Stemming from Telephone Conference Calls on September 6, 2006 and September 25, 2006, Request on October 4, 2006, and this Amendment.

LRA Section B.1.15, attribute 4, Detection of Aging Effects, is revised as follows (bold words added).

4. Detection of Aging Effects

Loss of material is the aging effect managed by this program. Representative tubes within the sample population of heat exchangers will be eddy current tested at a frequency determined by internal and external operating experience to ensure that effects of aging are identified prior to loss of intended function. Visual inspections of accessible heat exchangers will be performed on the same frequency as eddy current inspections.

An appropriate sample population of heat exchangers will be determined based on operating experience prior to inspections. **The sample population of heat exchangers will be determined based on the materials of construction of the heat exchanger tubes and the associated environments as well as the type of heat exchanger (for example, shell and tube type). At least one heat exchanger of each type, material and environment combination will be included in the sample population.** Inspection can reveal loss of material that could result in degradation of the heat exchangers. Fouling is not addressed by this program.

LRA Section B.1.22 is revised as follows (underlined words added, strike-outs deleted)

NUREG-1801 Consistency

The Oil Analysis Program at PNPS is consistent with the program described in NUREG-1801, Section XI.M39, Lubricating Oil Analysis, with ~~an exception~~ exceptions and enhancements.

Exceptions to NUREG-1801

The Oil Analysis Program at PNPS is consistent with the program described in NUREG-1801, Section XI.M39, Lubricating Oil Analysis with the following ~~exception~~ exceptions.

Attributes Affected	Exception
3. Parameters Monitored/Inspected	Flash point is not determined for sampled oil.[1]
3. Parameters Monitored/Inspected	Neutralization number and fuel dilution are not monitored for every oil sample.[2]

1. Analyses of filter residue or particle count, viscosity, total acid/base (neutralization number), water content, and metals content provide sufficient information to verify the oil is suitable for continued use.

2. The parameters monitored regularly (presence of moisture, abnormal wear products, and changes in viscosity) are those directly related to age-related degradation of components containing lube oil. As noted in the Mechanical Tools, aging effects are not observed in fuel oil and lubricating oil systems unless moisture or other contaminants are present. Therefore, continuous monitoring and trending of particle count, water content and viscosity in lubricating oil provides reasonable assurance that effects of aging will be managed such that applicable components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

The following LRA changes stem from a telephone conference call held with the NRC license renewal staff on September 6, 2006.

All fuse holders at Pilgrim are either part of a complex active assembly or part of circuits that perform no license renewal intended function. Therefore, LRA Section 2.5 discussion of exceptions to components subject to aging management review is revised to delete "with metallic clamps" following "fuse holders" in the final bullet on page 2.5-2.

Table 2.5-1 and Table 3.6.1-2 are revised to delete line items for fuse holders.

The following sentence is added to the discussion column of item 3.6.1-2 in Table 3.6.1.

> For fuse holders, see items 3.6.1-6 and 3.6.1-14.

The discussion column of item 3.6.1-6 in Table 3.6.1 is revised to state the following.

> NUREG-1801 aging effect is not applicable to PNPS. All fuse holders at PNPS are either part of a complex active assembly or part of circuits that perform no license renewal intended function. Therefore, fuse holders at PNPS are not subject to aging management review.

The discussion column of item 3.6.1-14 in Table 3.6.1 is revised to state the following.

> All fuse holders at PNPS are either part of a complex active assembly or part of circuits that perform no license renewal intended function. Therefore, fuse holders at PNPS are not subject to aging management review.

LRA Section 2.5 and Table 2.5-1 are further clarified by addition of the following statements.

> "Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements" includes electrical penetrations conductors and connections.
>
> "Switchyard bus" includes connections.

The following LRA changes and clarifications stem from a telephone conference call with the NRC license renewal staff on September 25, 2006.

Item 1 – Relief Requests

Since ASME code relief requests have their own process under 10 CFR 50.55a, reference to relief requests in the LRA is unnecessary. The following changes are made to the LRA to remove reference to relief requests (strike-outs deleted).

- Section B.1.16, page B-55, last sentence of fifth paragraph is revised as shown below.

  For containment inservice inspection, general visual and detailed visual examinations are used in addition to VT examinations as allowed by 10 CFR 50.55a ~~to include applicable relief requests~~.

- Section B.1.16.2, page B-59, first paragraph in Scope of Program is revised as shown below.

  The ISI Program manages cracking, loss of material, and reduction of fracture toughness of reactor coolant system piping, components, and supports. The program implements applicable requirements of ASME Section XI, Subsections IWA, IWB, IWC, IWD and IWF, and other requirements specified in 10 CFR 50.55a ~~with approved NRC alternatives and relief requests~~. Every 10 years the ISI Program is updated to the latest ASME Section XI code edition and addendum approved by the NRC in 10 CFR 50.55a.

Item 2 – Steam Dryer Inspections

In LRA Amendment 5, in response to Audit item 320, LRA Appendix A was revised to identify commitment numbers associated with new and enhanced programs. The revision of LRA Appendix A was amended in LRA Amendment 8 (September 13, 2006). The revision to LRA Appendix A is further revised as follows (bold words added, strike-outs deleted).

Section A.2.1.8, BWR Vessel Internals Program, add "License renewal commitments 3, ~~and~~ 33 **and 37** specify enhancements to this program."

Item 3 – Clarification of RAI 2.3.3.3-1 Response

A typo was noted in the response to RAI 2.3.3.3-1 provided in LRA Amendment 7 (August 30, 2006). The response is amended as follows (bold words added, strike-outs deleted).

Flexible hoses in the RBCCW system are replaced based on a specified time period and are therefore not subject to aging management review. Drawings LRA-M-**215**~~264~~ sheets 1, 2, and 4 incorrectly show flexible connections as being subject to aging management review.

Item 4 – Clarification of RAI 2.3.3.4-5 Response

The response to RAI 2.3.3.4-5 provided in LRA Amendment 7 (August 30, 2006) is clarified as follows (bold words added).

> The crankcase exhauster is not shown on the drawing because it is physically attached to the diesel engine block and is considered part of the diesel engine. In accordance with NEI 95-10 revision 6 Appendix B, emergency diesel engines do not meet 10 CFR 54.21(a)(1)(i) because they are active and are therefore not subject to aging management review. The effects of aging on components that are part of the active diesel engine are managed under the Maintenance Rule 10 CFR 50.65.
>
> **The "crankcase exhauster" labels on drawing LRA-M-272-0 indicate only that the jacket water pressure switches (PS-JWPS-4A, B) provide an engine running signal to the crankcase exhauster motors. They are not intended to imply that the crankcase exhausters are external to the engine. Each crankcase exhauster, driven by an electric motor, is a centrifugal blower which exhausts crankcase vapors to the atmosphere. The crankcase exhauster assembly is mounted on top of the cylinder block of the engine.**
>
> **The LRA Table 3.3.2-4 line items for carbon steel piping containing exhaust gas do not apply to the crankcase exhausters as they do not contain exhaust gas. These line items apply to the exhaust piping exiting the turbocharger.**

Item 5 – Clarification of RAI 2.3.3.4-3 Response

The response to RAI 2.3.3.4-3 provided in LRA Amendment 7 (August 30, 2006) is clarified as follows (bold words added).

> The turbocharger interface with the jacket water cooling system was inadvertently omitted from the LRA. The intended function of heat transfer is added to Table 2.3.3-4 for component type turbocharger **housing.** Table 3.3.2-4 is also revised to add additional line items for component type turbocharger **housing** as follows.

Turbo-charger **housing**	Pressure boundary	Carbon steel	Treated water > 140°F (int)	Loss of material	Water chemistry control- closed cooling water	VII.H2-23 (A-25)	3.3.1-47	D
Turbo-charger **housing**	Heat transfer	Carbon steel	Treated water > 140°F (int)	Fouling	Water chemistry control- closed cooling water	VII.F1-13 (AP-77)	3.3.1-52	D

<u>The following LRA changes and clarification on metal fatigue stem from a request from the NRC license renewal staff on October 4, 2006.</u>

LRA Amendment 5 (July 19, 2006) included commitments 31 and 35 to address metal fatigue. The commitments were subsequently revised in LRA Amendment 8 (September 13, 2006). However, revisions to the LRA Appendix A, UFSAR Supplement, subsections A.2.2.2.1 and A.2.2.2.3 to include these commitment numbers were inadvertently omitted.

LRA UFSAR Supplement subsections A.2.2.2.1 is revised to include the following statement.

> License renewal commitment 35 addresses metal fatigue for reactor vessel components, including the feedwater nozzles.

LRA UFSAR Supplement subsections A.2.2.2.3 is revised to include the following statement.

> License renewal commitment 31 addresses environmental-assisted fatigue for the locations identified in NUREG/CR-6260 for BWRs of the PNPS vintage.

The following LRA change stems from this amendment (LRA Amendment 9).

In LRA Amendment 5, in response to Audit item 320, LRA Appendix A was revised to identify commitment numbers associated with new and enhanced programs. The revision of LRA Appendix A was amended in LRA Amendment 8 (September 13, 2006). The revision to LRA Appendix A is further revised as follows (bold word and number added, strike-outs deleted).

> Section A.2.1.10, Diesel Fuel Monitoring Program, add "License renewal commitments 4, 5 ~~and~~ 6, **and 38** specify enhancement to this program."