From: Perry Buckberg
To: Christopher Sydnor; Dan Hoang; James Davis; Lambros Lois; Matthew Mitchell; Naeem IQBAL
Date: 5/1/2007 9:20:11 PM
Subject: Pilgrim Amendment

The attached files make up LRA Amendment 16 dated 5/1/2007. I will be out of the office tomorrow (5/2/2007) but will check on the status Thursday.

Thanks,
Perry

CC: Louise Lund

Mail Envelope Properties (4637E74B.8F6 : 9 : 8248)

Subject:	Pilgrim Amendment
Creation Date	5/1/2007 9:20:11 PM
From:	Perry Buckberg
Created By:	PHB1@nrc.gov

Recipients	Action	Date & Time
nrc.gov		
OWGWPO02.HQGWDO01	Delivered	5/1/2007 9:20:24 PM
NXI (Naeem IQBAL)		
nrc.gov		
OWGWPO03.HQGWDO01	Delivered	5/1/2007 9:20:17 PM
JAD (James Davis)		
nrc.gov		
OWGWPO04.HQGWDO01	Delivered	5/1/2007 9:20:21 PM
DVH (Dan Hoang)		
nrc.gov		
TWGWPO01.HQGWDO01	Delivered	5/1/2007 9:20:25 PM
LXL CC (Louise Lund)		
LXL1 (Lambros Lois)		
MAM4 (Matthew Mitchell)		
nrc.gov		
TWGWPO02.HQGWDO01	Delivered	5/1/2007 9:20:22 PM
CRS (Christopher Sydnor)		

Post Office	Delivered	Route
OWGWPO02.HQGWDO01	5/1/2007 9:20:24 PM	nrc.gov
OWGWPO03.HQGWDO01	5/1/2007 9:20:17 PM	nrc.gov
OWGWPO04.HQGWDO01	5/1/2007 9:20:21 PM	nrc.gov
TWGWPO01.HQGWDO01	5/1/2007 9:20:25 PM	nrc.gov
TWGWPO02.HQGWDO01	5/1/2007 9:20:22 PM	nrc.gov

Files	Size	Date & Time
MESSAGE	496	5/1/2007 9:20:11 PM
Pilgrim LRA Amendment 16 5_1_2007.pdf	2862546	5/1/2007 8:15:32 PM
Pilgrim LRA Amendment 16 2 5_1_2007.pdf	3201616	5/1/2007 8:18:46 PM

Options

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Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

Stephen J. Bethay
Director, Nuclear Assessment

May 1, 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 16

REFERENCES:
1. Entergy Letter, License Renewal Application, dated January 25, 2006
2. NRC Safety Evaluation Report with Open Items Related to the Pilgrim License Renewal Application, dated March 2007
3. NRC Request for additional information for review of the Pilgrim license renewal application, dated March 26, 2007
4. Entergy Letter, Comments on NRC Draft Safety Evaluation Report Related to PNPS LRA, dated March 28, 2007

LETTER NUMBER: 2.07.027

Dear Sir or Madam:

In Reference 1, Entergy Nuclear Operations, Inc. applied for renewal of the Pilgrim Nuclear Power Station operating license. NRC TAC No. MC9669 was assigned to the application.

This letter provides information to address the Open Items from the NRC safety evaluation report (SER), (Reference 2). This letter also provides information in response to a request for additional information (Reference 3) related to Open Item 4.2. In addition, this letter includes LRA amendments resulting from review of the NRC SER.

Commitments made by this letter are contained in Attachment A.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 1, 2007.

Sincerely,

Stephen J. Bethay

Stephen J. Bethay
Director Nuclear Safety Assessment

ERS/dl

Attachments:

Attachment A:	Revised List of Regulatory Commitments
Attachment B:	Information in Response to the Open Items Listed in the Draft NRC SER, Including Associated LRA Amendments and License Condition
Attachment C:	LRA Amendments to Delete the BWRVIP-48 and BWRVIP-49 Fatigue Assessments as TLAAs
Attachment D:	Torus Room Concrete Base Mat Evaluation (Dr. Franz Ulm, M.I.T.)
Attachment E:	Structural Integrity Associates Fluence Evaluation for PNPS

cc: see next page

cc: with Attachments

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ATTACHMENT A to Letter 2.07.027
(8 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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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 2.06.003 and 2.06.057 and 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 2.06.057 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	Letters 2.06.003 and 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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057 and 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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	B.1.22 / Audit Item 320
20	Implement the One-Time Inspection Program as described in LRA Section B.1.23.	June 8, 2012	Letters 2.06.003 and 2.06.057 and 2.07.023	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	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	Letters 2.06.003 and 2.06.057	B.1.30 / Audit Items 320, 327

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
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	Letters 2.06.003 and 2.06.057	B.1.31 / Audit Items 257, 320
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. The determination of Fen will account for operating times with both hydrogen water chemistry and normal water chemistry. (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 2.06.057 and 2.06.064 and 2.06.081 and 2.07.005	4.3.3 / Audit Items 302, 346

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
32	Implement the enhanced Bolting Integrity Program described in Attachment C of Pilgrim License Renewal Application Amendment 5 (Letter 2.06.064).	June 8, 2012	Letters 2.06.057 and 2.06.064 and 2.06.081	Audit items 364, 373, 389, 390, 432, 443, 470
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	Letters 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 2.06.057 and 2.06.064 and 2.06.081	Audit Item 345

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
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
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
39	Perform a one-time inspection of the Main Stack foundation prior to the period of extended operation.	June 8, 2012	Letter 2.06.094	B.1.23 / Audit Item 581
40	Enhance the Oil Analysis Program by documenting program elements 1 through 7 in controlled documents. The program elements will include enhancements identified in the PNPS license renewal application and subsequent amendments to the application. The program will include periodic sampling for the parameters specified under the Parameters Monitored/Inspected attribute of NUREG-1801 Section XI.M39, Lubricating Oil Analysis. The controlled documents will specify appropriate acceptance criteria and corrective actions in the event acceptance criteria are not met. The basis for acceptance criteria will be defined.	June 8, 2012	Letter 2.06.094	B.1.22 / Audit Items 553 and 589
41	Enhance the Containment Inservice Inspection (CII) Program 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.	June 8, 2012	Letter 2.06.094	A.2.1.17 and B.1.16.1
42	Implement the Bolted Cable Connections Program, described in Attachment C of Pilgrim License Renewal Application 11 (Letter 2.07.003), prior to the period of extended operation.	June 8, 2012	Letter 2.07.003	A.2.1.40 and B.1.34

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
43	Include within the Structures Monitoring Program provisions to ensure groundwater samples are evaluated periodically to assess the aggressiveness of groundwater to concrete, as described in Attachment E of LRA Amendment 12 (Letter 2.07.005), prior to the period of extended operation.	June 8, 2012	Letter 2.07.005	A.2.1.32 and B.1.29.2
44	Perform another set of the UT measurements 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.	As stated in the commitment.	Letter 2.07.010	A.2.1.17 and B.1.16.1
45	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 and once within the first ten years of the period of extended operation.	As stated in the commitment.	Letters 2.07.010 and 2.07.027	A.2.1.32 and B.1.29.2
46	Inspect the condition of a sample of the torus hold-down bolts and associated grout and determine appropriate actions based on the findings prior to the period of extended operation.	June 8, 2012	Letter 2.07.027	A.2.1.32 and B.1.29.2
47	Submit to the NRC an action plan to improve benchmarking data to support approval of new P-T curves for Pilgrim.	Sept.15, 2007	Letter 2.07.027	4.2.2, A.2.2.1.1, and A.2.2.1.2
48	On or before June 8, 2010, Entergy will submit to the NRC calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.	June 8, 2010	Letter 2.07.027	4.2, 4.7.1, A.1.1 and A.2.2.1

ATTACHMENT B to Letter 2.07.027

(17 pages)

Information in Response to the Open Items Listed in the Draft NRC SER,
Including Associated LRA Amendments and License Condition

<u>OI 2.3.3.6:</u> <u>(SER Section 2.3.3.6 - Security Diesel)</u>

LRA Table 2.3.3-6 shows the component types subject to an AMR but the security diesel system was not in the FSAR or in any license renewal drawings; therefore, the staff could not determine the portion of the security diesel system within the scope of license renewal. Additionally, the staff could not determine whether any components within the scope of license renewal were not shown as subject to an AMR. The staff referred this issue to NRC Region I who will determine whether security diesel system components are within the scope of license renewal.

<u>OI 2.3.3.6 Response</u>

Entergy provided NRC Region I with support as requested.

OI 3.0.3.2.10: (SER Section 3.0.3.2.10 - Fire Protection Program)

The applicant is taking an exception to the GALL Report program element "detection of aging effects," specifically:

The NUREG-1801 program states that approximately 10 percent of each type of penetration seal should be visually inspected at least once every refueling outage. The PNPS program specifies inspection of approximately 20 percent of the seals each operating cycle, with all accessible fire barrier penetration seals being inspected at least once every five operating cycles.

The LRA states that, because aging effects typically are manifested over several years, this variation in inspection frequency is insignificant. GALL AMP XI.M26 specifies approximately 10 percent of each type of seal should be inspected visually at least every refueling outage (two years). The applicant clarified that the program specifies inspection of approximately 20 percent of the seals, including at least one seal of each type, each operating cycle, with all accessible fire barrier penetration seals being inspected at least once every five operating cycles. The applicant needs to address how to manage the aging effect of inaccessible fire barrier penetration seals.

OI 3.0.3.2.10 Response

The PNPS requirement to inspect penetration seals applies to 100% of the seals. The word "accessible" is not necessary in the discussion of the exception for Detection of Aging Effects in the PNPS program. All fire barrier penetration seals are inspected at least once every five operating cycles. In LRA Appendix B, Section B.1.13.1, the word "accessible" is removed resulting in the following description of the exception for Detection of Aging Effects.

> The NUREG-1801 program states that approximately 10% of each type of penetration seal should be visually inspected at least once every refueling outage. The PNPS program specifies inspection of approximately 20% of the seals each operating cycle, with all ~~accessible~~ fire barrier penetration seals being inspected at least once every five operating cycles.

OI 3.0.3.3.2: <u>(SER Section 3.0.3.3.2 - Containment Inservice Inspection and Section 3.5.2.2.1 - PWR and BWR Containments)</u>

A recent NRC Region 1 inspection team observations indicated the following:

- 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 failed in 1999.
- Monitoring of other drains has been inconclusive and not well documented.
- The torus room floor has had water on the floor on multiple occasions.

In Request for Additional Information (RAI) B.1.16.1, dated November 7, 2006, the applicant was asked to address the above finding and discuss the 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, basemat, including the sand pocket region for the period of extended operation.

OI 3.0.3.3.2 Response

Entergy letter dated March 13, 2007 provided information to address this open item and RAI B.1.16.1. With regard to the issue of water on the torus room floor, Attachment D to this letter contains a report prepared by a consultant to Entergy that provides a detailed evaluation of the groundwater seepage through the concrete basemat.

Commitments 43, 45, and 46 will be implemented to address this issue. Commitment 45 made in the March 13, 2007 Entergy letter is revised by this letter to require it be performed once within the first ten years of the period of extended operation in addition to it being performed once prior to the period of extended operation. Commitment 46 is added by this letter. These commitments are listed in Attachment A to this letter and read as follows:

43	Include within the Structures Monitoring Program provisions to ensure groundwater samples are evaluated periodically to assess the aggressiveness of groundwater to concrete, as described in Attachment E of LRA Amendment 12 (Letter 2.07.005), prior to the period of extended operation.
45	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 and once within the first ten years of the period of extended operation.
46	Inspect the condition of a sample of the torus hold-down bolts and associated grout and determine appropriate actions based on the findings prior to the period of extended operation.

OI 4.2: (SER Sections: 3.0.3.2.15 - Reactor Vessel Surveillance Program, 4.2 - Reactor Vessel Neutron Embrittlement, 4.7.1 - Reflood Thermal Shock of the Reactor Vessel Internals, 4.7.2.1 BWRVIP-05, Reactor Vessel Circumferential Welds)

Due to the lack of benchmarking data in support of the plant-specific RAMA fluence calculations, the staff finds neutron fluence values unacceptable for use in the reactor vessel neutron embrittlement TLAAs.

OI 4.2 Response

OI 4.2 was clarified by the NRC in a request for additional information (RAI) transmitted in a letter dated March 26, 2007. The RAIs and responses are provided below.

RAI# 4.2

1. Fluence was calculated for the Pilgrim reactor vessel (RV) for the extended 60-year licensed operating period (54 effective full power years *(EFPY)* of facility operation), using the Radiation Analysis Modeling Application (RAMA) fluence methodology. The RAMA fluence methodology was previously approved by the NRC staff, and the results are acceptable for licensing actions provided that: (1) the RAMA application follows the guidance in Regulatory Guide 1.190 and (2) RV fluence calculations have at least one credible plant-specific surveillance capsule for benchmarking.

   The applicant provided 54 EFPY fluence values for the Pilgrim RV beltline materials in Section 4.2.1 of the License Renewal Application (LRA). These fluence values were used throughout Section 4.2 of the LRA for the RV neutron embrittlement time limited aging analyses (TLAAs). However, due to the lack of a credible plant-specific benchmark, the staff finds the 54 EFPY fluence values provided in LRA Section 4.2.1 unacceptable for use in the RV neutron embrittlement TLAAs. Therefore, the staff requests that the applicant revise Section 4.2.1 of the LRA to provide an acceptable neutron fluence evaluation or an alternative proposal for closing this TLAA topic in the LRA review.

2. Due to the lack of benchmarking data in support of the plant-specific RAMA fluence calculations, the staff cannot complete its review of the TLAAs in LRA Sections 4.2.2, 4.2.3, 4.2.4, 4.2.5, 4.2.6 and 4.7.1, as well as the aging management program (AMP) on the RV material surveillance program, using the current fluence values for the Pilgrim RV that were provided in LRA Section 4.2.1. Therefore, the staff requests that the applicant revise LRA Sections 4.2.2, 4.2.3,4.2.4, 4.2.5, 4.2.6, 4.7.1, and the AMP on the RV material surveillance program to provide an acceptable evaluation of these topics or an alternative proposal for closing these topics in the LRA review.

Response

The benchmarking validation of the RAMA fluence calculation is ongoing for the Pilgrim reactor vessel and internals. The RAMA calculated fluence is approximately 56% of the benchmark fluence calculated from the available surveillance capsule dosimetry. Uncertainties between the calculated and measured results from the dosimetry are still being examined to determine a possible cause for the discrepancy. To ensure resolution of this issue, Commitment 47, which reads as follows, is added by this letter.

47	On or before September 15, 2007 submit to the NRC an action plan to improve benchmarking data to support approval of new P-T curves for Pilgrim.

To address this issue, an alternative analysis is provided as a means to close this TLAA topic in the LRA review. To address fluence-related TLAAs for the period of extended operation, Entergy has evaluated the affected TLAAs to determine the limiting fluence value. The evaluation included information presented in LRA sections 4.2.1, 4.2.2, 4.2.3, 4.2.4, 4.2.5, 4.2.6, 4.7.1, and the AMP on the RV material surveillance program. From this evaluation the limiting fluence was determined.

The alternative analysis to determine the limiting fluence value is included as Attachment E. This analysis assumes increasing fluence levels until an ASME Code or regulatory limit is reached based on the projected changes in material properties. Changes in the vessel (ferritic) steel material properties are measured by an increase in adjusted reference temperature or a decrease in Charpy upper shelf energy. The effects of increasing fluence on the austenitic stainless steel core shroud and internals was also considered. By assuming increasing fluence levels, the analysis identifies the maximum fluence that can be experienced while meeting the Code and regulatory criteria. This analysis also shows that there is a large margin available to this limiting fluence at the end of the period of extended operation.

The analysis determined that the limiting fluence value was set by the temperature required to perform the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400 that requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed increasing fluence and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum starting temperature for the hydrostatic test was set at 212°F with a corresponding maximum allowable 1/4T fluence of 4.12E+18 n/cm^2.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for all fluence-related TLAAs. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that all of the fluence-related TLAAs remain valid, Commitment 48, which reads as follows, is added by this letter.

48	On or before June 8, 2010, Entergy will submit to the NRC calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

Entergy would find it acceptable if this commitment became a license condition.

It should be noted that at the ACRS meeting on April 4, 2007, reference was made to EPRI research that investigated the irradiated behavior of stainless steel components in order to predict service life. Further review has shown that the predictions of service life related to fluence are not directly relevant in this case. The core shroud and the top guide are components that are susceptible to aging effects. However, a review of the analyses related to the core shroud found that the only time-limited aging analysis (TLAA) involves the fatigue analysis and calculation of cumulative usage factors (CUFs) for the shroud repair. The core shroud does not affect the operating P-T limit curves and there is no criterion on fluence that would further limit the operation of the core shroud structure. Similarly, the top guide does not affect the operating P-T limit curves, and there is no criterion on fluence that would further limit the operation of the top guide structure.

PNPS has re-evaluated the neutron embrittlement issues of Sections 4.2 and 4.7.1 and prepared revised LRA sections below. The Reactor Vessel Material Surveillance Program, with the changes to the fluence extrapolation, is correct as written, and no changes to Appendix B, Section B.1.26 are necessary.

LRA Amendments

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The regulations governing reactor vessel integrity are in 10 CFR 50. Section 50.60 requires that all light-water reactors meet the fracture toughness, pressure-temperature limits, and material surveillance program requirements for the reactor coolant pressure boundary as set forth in 10 CFR 50 Appendices G and H.

The PNPS current licensing basis analyses evaluating reduction of fracture toughness of the PNPS reactor vessel for 40 years are TLAA. The reactor vessel neutron embrittlement TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii) as summarized below. Fifty-four effective full-power years (EFPY) are projected for the end of the period of extended operation (60 years) assuming an average capacity factor of 90% for 60 years.

4.2.1 Reactor Vessel Fluence

Calculated fluence is based on a time-limited assumption defined by the operating term. As such, fluence is the time-limited assumption for the time-limited aging analyses that evaluate reactor vessel neutron embrittlement.

Fluence values were calculated using the RAMA fluence methodology. The RAMA fluence methodology was developed for the Electric Power Research Institute, Inc. and the boiling water reactor vessel and internals project (BWRVIP) for the purpose of calculating neutron fluence in boiling water reactor components. This methodology has been approved by the NRC (Reference 4.2-20) for application in accordance with Regulatory Guide (RG) 1.190; assuming the results are appropriately benchmarked.

The benchmarking validation of the RAMA fluence calculation is ongoing for the Pilgrim reactor vessel. The RAMA calculated fluence is approximately 56% of the benchmark fluence calculated from the available surveillance capsule dosimetry. Uncertainties between the calculated and measured results from the dosimetry are still being examined to determine a possible cause for the discrepancy. Commitment 47 requires a plan for resolving this discrepancy to be developed and submitted for review by September 2007.

An alternative analysis to determine the limiting fluence value has been performed. This analysis assumes increasing fluence levels until an ASME Code or regulatory limit is reached based on the projected changes in material properties. Changes in the vessel (ferritic) steel material properties are measured by an increase in adjusted reference temperature or a decrease in Charpy upper shelf energy. The effects of increasing fluence on the austenitic stainless steel core shroud and internals was also considered. By assuming increasing fluence levels, the analysis identifies the maximum fluence that can be experienced while meeting the Code and regulatory criteria.

The analysis determined that the limiting fluence value was set by the temperature required to perform the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature starting for the hydrostatic test was set at 212°F The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E+18 n/cm^2. This fluence level was the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for all fluence-related TLAAs. Commitment 48 is to confirm that the limiting fluence will not be reached during the period of extended operation and consequently that all of the fluence-related TLAAs will be valid to the end of the period of extended operation.

At PNPS, the limiting beltline material for 40 years consists of 6 plates and their connecting welds, all adjacent to the active fuel zone. No nozzles are included in the limiting beltline materials for the current term of operation (Reference 4.2-2).

The beltline will be re-evaluated for 60 years. An evaluation of the RTNDT for nozzle forgings and welds is expected to show that their adjusted reference temperature at 54 EFPY will be well below the adjusted reference temperatures used in determining the P-T limits. Thus, the nozzle forgings and welds are not expected to be the limiting items for the period of extended operation.

4.2.2 Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires that reactor vessel boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences be accomplished within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the materials and fluence data obtained through the Reactor Vessel Surveillance Program.

Pilgrim received License Amendment 227 dated March 29, 2007 that extended the existing P-T limit curves for Pilgrim through Cycle 18.

The P-T limit curves will continue to be updated, as required by Appendix G of 10 CFR Part 50 or as operational needs dictate. This updating will assure that the operational limits remain valid through the period of extended operation. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

4.2.3 Charpy Upper-Shelf Energy

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy ... of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb...." The initial (unirradiated) values of upper-shelf energy (CVUSE) for PNPS beltline welds were provided to the NRC in correspondence responding to Generic Letter 92-01 (References 4.2-9, 4.2-10).

Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Revision 2, provides two methods for determining Charpy upper-shelf energy (CVUSE). Position 1 applies for material that does not have surveillance data and Position 2 applies for material with surveillance data. Position 2 requires a minimum of two sets of credible material surveillance data. Since PNPS has data from only one material surveillance capsule, Position 2 does not apply. For Position 1, the percent drop in CVUSE for a stated copper content and neutron fluence is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial CVUSE to obtain the adjusted CVUSE.

The predictions for percent drop in CVUSE at 54 EFPY must be based on chemistry data, the maximum 1/4T fluence values, and unirradiated CVUSE data submitted to the NRC in the PNPS response to GL 92-01. The predicted CVUSE values for 54 EFPY will utilize Regulatory Guide 1.99 Position 1. The predictions will use Regulatory Guide 1.99, Position 1, Figure 2; specifically, the formula for the lines will be used to calculate the percent drop in CVUSE (Reference 4.2-14).

PNPS will use chemistry data from previous licensing submittals, the PNPS response to GL 92-01 (References 4.2-9, 4.2-10, 4.2-14), and the 1/4T fluence values to be determined to perform linear interpolation on the CVUSE percent drop values in RG 1.99, Revision 2, Figure 2.

The license renewal SER for BWRVIP-74 (Reference 4.2-11), Action Item #10, states that each license renewal applicant shall demonstrate that the percent reduction in Charpy USE for their beltline materials is less than that specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds given in BWRVIP-74. This action item is not applicable to PNPS if the PNPS projected CVUSE remains above the 50 ft-lb limit, even for the period of extended operation.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel Charpy upper shelf energy TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

4.2.4 Adjusted Reference Temperature

Irradiation by high-energy neutrons raises the value of RT_{NDT} for the reactor vessel. RT_{NDT} is the reference temperature for nil-ductility transition as defined in Section NB-2320 of the ASME Code. The initial RT_{NDT} is determined through testing of unirradiated material specimens. The shift in reference temperature, ΔRT_{NDT}, is the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. The adjusted reference temperature (ART) is defined as initial RT_{NDT} + ΔRT_{NDT} + margin. The margin is defined in RG 1.99, Revision 2. The P-T curves are developed from the ART value for the vessel materials. RG 1.99 Revision 2 defines the calculation methods for RT_{NDT} and ART.

The PNPS reactor vessel was evaluated for an assumed exposure of less than 10^{19} nvt of neutrons with energies exceeding 1 MeV (Reference 4.2-1). After approximately 4.17 EFPY, the first surveillance capsule was withdrawn from the vessel and tested. The capsule test report concludes that the shift in RT_{NDT} and upper-shelf energy over 32 EFPY will be within 10 CFR 50 guidelines.

PNPS will project values for ΔRT_{NDT} and ART at 54 EFPY using the methodology of RG 1.99. These values will be calculated using the chemistry data, margin values, initial RT_{NDT} values, and chemistry factors (CFs) contained in the PNPS response to GL 92-01 (References 4.2-3, 4.2-9, 4.2-10, 4.2-13). Initial RT_{NDT} values are from report SIR-00-082, which was submitted in 2001 as part of the PNPS P-T limit change request (Reference 4.2-5). The 1/4T fluence values discussed in Section 4.2.1 will be used. New fluence factors (FFs) will be calculated using the expression in RG 1.99, Revision 2, Equation 2, where the fluence factor is given by

$$FF = f^{(0.28-0.10*\log f)}$$

In this equation, *f* is the 1/4T fluence value. The new ΔRT_{NDT} values will be calculated by multiplying the CF and the FF for each plate and weld. Calculated margins and the initial RT_{NDT} will then be added to the calculated ΔRT_{NDT} in order to arrive at the new value of ART.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel adjusted reference temperature TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

4.2.5 Reactor Vessel Circumferential Weld Inspection Relief

Relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 is based on an analysis indicating acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

PNPS received NRC approval for this relief for the remainder of the original 40-year license term. The basis for this relief request is an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement (References 4.2-16, 4.2-17). The anticipated changes in metallurgical conditions expected over the extended operating period require additional analysis to extend this relief request.

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the reactor pressure vessel (RPV) shell weld failure probabilities. Three key inputs to the PFM analysis are (1) the estimated end-of-life mean neutron fluence, (2) mean chemistry values based on vessel types, and (3) the assumption of potential for beyond-design-basis events.

PNPS will compare the reactor vessel limiting circumferential weld parameters to those used in the NRC analysis for the first two key assumptions. The data will be from the NRC SER for PNPS Relief Request 28 (Reference 4.2-17), and from the data in Table 2.6.4 of the NRC SER for BWRVIP-05 (Reference 4.2-18). (For comparison, the EOL mean RT_{NDT} will be calculated without margin and hence will be lower than the Section 4.2.2 RT_{NDT} value.)

The procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when PNPS requested approval of the BWRVIP-05 technical alternative for the current license term.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel circumferential weld failure probability TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

4.2.6 Reactor Vessel Axial Weld Failure Probability

The BWRVIP recommendations for inspection of reactor vessel shell welds (BWRVIP-05) are based on generic analyses supporting an NRC SER conclusion that the generic-plant axial weld failure rate is no more than 5×10^{-6} per reactor year (Reference 4.2-18). BWRVIP-05 showed that this axial weld failure rate is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above.

PNPS received relief from the circumferential weld inspections for the remainder of the original 40-year operating term (Reference 4.2-17). The basis for this relief request was a plant-specific analysis that showed the limiting conditional failure probability for the PNPS circumferential welds at the end of the original operating term was less than the values calculated in the BWRVIP-05 SER (Reference 4.2-11). The BWRVIP-05 SER concluded that the reactor vessel failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than 5×10^{-6} per reactor year. This failure frequency is dependent upon given assumptions of flaw density, distribution, and location. The failure frequency also assumes that "essentially 100%" of the reactor vessel axial welds will be inspected. The PNPS relief request requires additional relief request if less than 90% coverage is achieved.

PNPS will compare the reactor vessel limiting axial weld parameters to those used in the NRC analysis. The parameters used will be those from the NRC SER for BWRVIP-05 (Reference 4.2-18) from the NRC Supplemental SER for BWRVIP-05 (Reference 4.2-19).

The supplemental SER required the limiting axial weld to be compared with data found in Table 3 of the document. Originally, the supplemental SER identified PNPS as a limiting plant for the BWR fleet; however, in the discussion it is noted that the high EOL value of RT_{NDT} for PNPS calculated by the BWRVIP is due to the use of an initial RT_{NDT} of 0°F. The supplemental SER notes that the docketed value of initial RT_{NDT} (from the RVID) is -48°F, and therefore the EOL value of RT_{NDT} for PNPS is not bounding for the BWR fleet. The supplemental SER stated that the axial welds for the Clinton plant are the limiting welds for the BWR fleet and vessel failure probability determined for Clinton should bound the BWR fleet.

The limiting values will be compared to the values assumed in the analysis performed by the NRC staff in the BWRVIP-05 supplemental SER and the 64 EFPY limits and values obtained from Table 2.6- 5 of the SER. As such, this TLAA will be projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel axial weld failure probability TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

4.7.1 Reflood Thermal Shock of the Reactor Vessel Internals

UFSAR Section 3.3.6.8 addresses reflood thermal shock of the reactor vessel internals (core shroud). This evaluation of thermal shock was considered a TLAA as it is potentially based on shroud material properties that are affected by neutron fluence.

The shroud material is Type 304 stainless steel, which is not significantly affected by irradiation.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence level was the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reflood thermal shock TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

<u>Changes to existing UFSAR Section 3.3.6.8 information presented in Section A.1.1 of the LRA (page A-3) are revised as follows:</u>

<u>3. Shroud inner surfaces at highest irradiation zone</u>

~~The most irradiated point on the inner surface of the shroud is subjected to a total integrated neutron flux of 2.7 x 10^{20} nvt (>1 MeV) by the end of station life.~~ The peak thermal shock stress is 155,700 psi, corresponding to a peak strain of 0.57 percent. The shroud material is Type 304 stainless steel, which is not significantly affected by irradiation. The material does experience a loss in reduction of area. Because reduction of area is the property which determines tolerable local strain, irradiation effects can be neglected. The peak strain resulting from thermal shock at the inside of the shroud represents no loss of integrity of the reactor vessel inner volume. ~~The service limit of Type 304 stainless steel is approached at a fluence of 8 x 1021 n/cm2 (BWRVIP-35). As the PNPS shroud will remain below that fluence level for the period of extended operation, the shroud will remain serviceable.~~

<u>UFSAR Supplement Sections are revised to read as follows:</u>

A.2.2.1.1 Reactor Vessel Fluence

Calculated fluence is based on a time-limited assumption defined by the operating term. As such, fluence is the time-limited assumption for the time-limited aging analyses that evaluate reactor vessel embrittlement. Fluence values were calculated using the RAMA fluence calculation method. The RAMA fluence method was developed for the Electric Power Research Institute, Inc. and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for the purpose of calculating neutron fluence in boiling water reactor components. This method has been approved by the NRC (Reference A.2-9) for application in accordance with Regulatory Guide 1.190 provided the fluence calculations for the reactor are appropriately benchmarked.

The benchmarking validation of the RAMA fluence calculation is ongoing for the PNPS reactor vessel. The RAMA calculated fluence is approximately 56% of the benchmark fluence calculated from the available surveillance capsule dosimetry. Uncertainties between the calculated and measured results from the dosimetry are still being examined to determine a possible cause for the discrepancy. An action plan to improve benchmarking data to support approval of new P-T curves will be developed and submitted for NRC review.

An alternative analysis to determine the limiting fluence value has been performed (Reference A.2-12). This analysis assumes increasing fluence levels until an ASME Code or regulatory limit is reached based on the projected changes in material properties. Changes in the vessel (ferritic) steel material properties are measured by an increase in adjusted reference temperature or a decrease in Charpy upper shelf energy. The effects of increasing fluence on the austenitic stainless steel core shroud and internals was also considered. By assuming increasing fluence levels, the analysis identifies the maximum fluence that can be experienced while meeting the Code and regulatory criteria.

The analysis determined that the limiting fluence value was set by the temperature required to perform the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature starting for the hydrostatic test was set at 212°F The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E+18 n/cm^2. This fluence level was the limiting fluence value identified.

On or before June 8, 2010, Entergy will submit to the NRC calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

A.2.2.1.2 Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires that reactor vessel boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences be accomplished within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the materials and fluence data obtained through the Reactor Vessel Surveillance Program.

Pilgrim received License Amendment 227 dated March 29, 2007 that extended the existing P-T limit curves for Pilgrim through Cycle 18.

The P-T limit curves will continue to be updated, as required by Appendix G of 10 CFR Part 50 or as operational needs dictate. This updating will assure that the operational limits remain valid through the period of extended operation. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

A.2.2.1.3 Charpy Upper-Shelf Energy

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy ... of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb...." The initial (unirradiated) values of upper-shelf energy (CvUSE) for PNPS beltline welds were provided to the NRC in correspondence responding to Generic Letter 92-01.

Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Revision 2, provides two methods for determining Charpy upper-shelf energy (CvUSE). Position 1 applies for material that does not have surveillance data and Position 2 applies for material with surveillance data. Position 2 requires a minimum of two sets of credible material surveillance data. Since PNPS has data from only one material surveillance capsule, Position 2 does not apply. For Position 1, the percent drop in CvUSE for a stated copper content and neutron fluence is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial CvUSE to obtain the adjusted CvUSE.

The predictions for percent drop in CVUSE at 54 EFPY must be based on chemistry data, the maximum 1/4T fluence values, and unirradiated CvUSE data submitted to the NRC in the PNPS response to GL 92-01. The predicted CvUSE values for 54 EFPY will utilize Regulatory Guide 1.99 Position 1. The predictions will use Regulatory Guide 1.99, Position 1, Figure 2; specifically, the formula for the lines will be used to calculate the percent drop in CvUSE.

PNPS will use chemistry data from previous licensing submittals, the PNPS response to GL 92-01, and the 1/4T fluence values to be determined to perform linear interpolation on the CvUSE percent drop values in RG 1.99, Revision 2, Figure 2.

The license renewal SER for BWRVIP-74, Action Item #10, states that each license renewal applicant shall demonstrate that the percent reduction in Charpy USE for their beltline materials is less than that specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds given in BWRVIP-74. This action item is not applicable to PNPS if the PNPS projected CvUSE remains above the 50 ft-lb limit, even for the period of extended operation.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel Charpy upper shelf energy TLAA. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy will submit to the NRC on or before June 8, 2010 calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

A.2.2.1.4 Adjusted Reference Temperature

Irradiation by high-energy neutrons raises the value of RT_{NDT} for the reactor vessel. RT_{NDT} is the reference temperature for nil-ductility transition as defined in Section NB-2320 of the ASME Code. The initial RT_{NDT} is determined through testing of unirradiated material specimens. The shift in reference temperature, ΔRT_{NDT}, is the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. The adjusted reference temperature (ART) is defined as initial RT_{NDT} + ΔRT_{NDT} + margin. The margin is defined in RG 1.99, Revision 2. The P-T curves are developed from the ART value for the vessel materials. RG 1.99 Revision 2 defines the calculation methods for RT_{NDT} and ART.

The PNPS reactor vessel was evaluated for an assumed exposure of less than 10^{19} nvt of neutrons with energies exceeding 1 MeV. After approximately 4.17 EFPY, the first surveillance capsule was withdrawn from the vessel and tested. The capsule test report concludes that the shift in RT_{NDT} and upper-shelf energy over 32 EFPY will be within 10 CFR 50 guidelines.

PNPS will project values for ΔRT_{NDT} and ART at 54 EFPY using the methodology of RG 1.99. These values will be calculated using the chemistry data, margin values, initial RT_{NDT} values, and chemistry factors (CFs) contained in the PNPS response to GL 92-01. Initial RT_{NDT} values are from report SIR-00-082, which was submitted in 2001 as part of the PNPS P-T limit change request. The 1/4T fluence values discussed in Section 4.2.1 will be used. New fluence factors (FFs) will be calculated using the expression in RG 1.99, Revision 2, Equation 2, where the fluence factor is given by

$$FF = f^{(0.28-0.10*\log f)}$$

In this equation, f is the 1/4T fluence value. The new ΔRT_{NDT} values will be calculated by multiplying the CF and the FF for each plate and weld. Calculated margins and the initial RT_{NDT} will then be added to the calculated ΔRT_{NDT} in order to arrive at the new value of ART.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel adjusted reference temperature TLAA. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy will submit to the NRC on or before June 8, 2010 calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

A.2.2.1.5 Reactor Vessel Circumferential Weld Inspection Relief

Relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 is based on an analysis indicating acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

PNPS received NRC approval for this relief for the remainder of the original 40-year license term. The basis for this relief request is an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement. The anticipated changes in metallurgical conditions expected over the extended operating period require additional analysis to extend this relief request.

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the reactor pressure vessel (RPV) shell weld failure probabilities. Three key inputs to the PFM analysis are (1) the estimated end-of-life mean neutron fluence, (2) mean chemistry values based on vessel types, and (3) the assumption of potential for beyond-design-basis events.

PNPS will compare the reactor vessel limiting circumferential weld parameters to those used in the NRC analysis for the first two key assumptions. The data will be from the NRC SER for PNPS Relief Request 28, and from the data in Table 2.6.4 of the NRC SER for BWRVIP-05. (For comparison, the EOL mean RT_{NDT} will be calculated without margin and hence will be lower than the Section 4.2.2 RT_{NDT} value.)

The procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when PNPS requested approval of the BWRVIP-05 technical alternative for the current license term.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel circumferential weld failure probability TLAA. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy will submit to the NRC on or before June 8, 2010 calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

A.2.2.1.6 Reactor Vessel Axial Weld Failure Probability

The BWRVIP recommendations for inspection of reactor vessel shell welds (BWRVIP-05) are based on generic analyses supporting an NRC SER conclusion that the generic-plant axial weld failure rate is no more than 5×10^{-6} per reactor year. BWRVIP-05 showed that this axial weld failure rate is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above.

PNPS received relief from the circumferential weld inspections for the remainder of the original 40-year operating term. The basis for this relief request was a plant-specific analysis that showed the limiting conditional failure probability for the PNPS circumferential welds at the end of the original operating term was less than the values calculated in the BWRVIP-05 SER. The BWRVIP-05 SER concluded that the reactor vessel failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than 5×10^{-6} per reactor year. This failure frequency is dependent upon given assumptions of flaw density, distribution, and location. The failure frequency also assumes that "essentially 100%" of the reactor vessel axial welds will be inspected. The PNPS relief request requires additional relief request if less than 90% coverage is achieved.

PNPS will compare the reactor vessel limiting axial weld parameters to those used in the NRC analysis. The parameters used will be those from the NRC SER for BWRVIP-05 from the NRC Supplemental SER for BWRVIP-05.

The supplemental SER required the limiting axial weld to be compared with data found in Table 3 of the document. Originally, the supplemental SER identified PNPS as a limiting plant for the BWR fleet; however, in the discussion it is noted that the high EOL value of RT_{NDT} for PNPS calculated by the BWRVIP is due to the use of an initial RT_{NDT} of 0°F. The supplemental SER notes that the docketed value of initial RT_{NDT} (from the RVID) is -48°F, and therefore the EOL value of RT_{NDT} for PNPS is not bounding for the BWR fleet. The supplemental SER stated that the axial welds for the Clinton plant are the limiting welds for the BWR fleet and vessel failure probability determined for Clinton should bound the BWR fleet.

The limiting values will be compared to the values assumed in the analysis performed by the NRC staff in the BWRVIP-05 supplemental SER and the 64 EFPY limits and values obtained from Table 2.6- 5 of the SER. As such, this TLAA will be projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel axial weld failure probability TLAA. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy will submit to the NRC on or before June 8, 2010 calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

<u>The following reference is added to UFSAR Supplement Section A.2.3.</u>

A.2-12 Bethay, Stephen J. (Entergy), to Document Control Desk (NRC), "License Renewal Application Amendment 16," letter 2.07.027 dated May 1, 2007, Attachment E, Structural Integrity Associates Fluence Evaluation for PNPS.

ATTACHMENT C to Letter 2.07.027
(1 page)

LRA Amendments to Delete the BWRVIP-48 and BWRVIP-49
Fatigue Assessments as TLAAs

LRA Amendments to Delete BWRVIP-48 and BWRVIP-49 Fatigue Assessments as TLAAs

Sections 4.7.2.2 and 4.7.2.3 of the draft SER discuss BWRVIP-48, Vessel ID Attachment Welds and BWRVIP-49, Instrument Penetrations, respectively. Both sections appropriately conclude that the BWRVIP analysis does not constitute a TLAA for license renewal since the CLB does not include a plant-specific 40-year CUF calculation for these components.

Section 4.7.2.3 states, "In a letter dated May 11, 2006, the applicant amended the LRA to delete the BWRVIP-49 fatigue assessment for the RPV IPNs as a TLAA for the LRA. The amendment deleted LRA Section A.2.2.6." The referenced amendment letter does not contain the stated change. Apparently the change was made to the Vermont Yankee LRA, but due to administrative oversight was not made to the PNPS LRA. The change is equally applicable to PNPS. Amendments to the LRA to delete the BWRVIP-49 fatigue assessment as a TLAA are provided.

Also, Section 4.7.2.2 should have a similar discussion of changes to the LRA necessary to delete discussion of BWRVIP-48 as a TLAA. Amendments to the LRA to delete the BWRVIP-48 fatigue assessment as a TLAA are provided.

<u>LRA Amendments to Delete the BWRVIP-48 Fatigue Assessment as a TLAA</u>

Delete Section 4.7.2.2, "BWRVIP-48, Vessel ID Attachment Welds."

Delete entry for BWRVIP-48, vessel ID attachment welds fatigue analysis from Table 4.1-1, "List of PNPS TLAA and Resolution."

Delete entry for cracking – fatigue with TLAA-metal fatigue from ID attachment welds in Table 3.1.2-1. Cracking managed by the BWR Vessel ID Attachment Welds Program remains in the table.

Delete Section A.2.2.5, "Vessel ID Attachment Welds Fatigue Analysis."

<u>LRA Amendments to Delete the BWRVIP-49 Fatigue Assessment as a TLAA</u>

Delete Section 4.7.2.3, "BWRVIP-49, Instrument Penetrations."

Delete entry for BWRVIP-49, instrument penetrations fatigue analysis from Table 4.1-1, "List of PNPS TLAA and Resolution."

Delete entry for cracking – fatigue with TLAA-metal fatigue from nozzles, reactor vessel instrumentation (N15, N16) in Table 3.1.2-1. Cracking managed by the BWR Penetrations Program remains in the table.

Delete Section A.2.2.6, "Instrument Penetrations Fatigue Analysis."

ATTACHMENT D to Letter 2.07.027

(9 pages)

Torus Room Concrete Base Mat Evaluation (Dr. Franz Ulm, M.I.T.)

Preliminary Durability Performance Evaluation of Torus Base Mat in Pilgrim Station

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Date: April 30, 2007

Executive Summary: The objective of this report is a preliminary durability evaluation of the effect of the observed groundwater intrusion on the structural performance of the 8 ft thick Reactor Building base mat in Entergy's Pilgrim Station. Based on observations and documents shared by the Entergy team with the author, the following conclusions are drawn:

1. The groundwater migration through the 8ft. thick Reactor Building base mat is a highly localized phenomenon. It is caused by a 25ft hydraulic head difference, pushing groundwater through vertical joints and zones most likely weakened by tensions generated during setting and hydration following the construction. These localized zones are discontinuities equivalent to a vertical cylindrical hole of a maximum diameter of 4 mm (1/6th in). Such small discontinuities that originate from construction joints are inevitable in large-scale concrete engineering operations.

2. This highly localized nature of the zones through which water penetrates, does not compromise the overall structural performance of the Torus base mat: it does neither affect the bulk integrity of the concrete slab, nor the overall compressive and bending load bearing capacity of the reactor foundation.

3. Calcium leaching of the solid concrete is expected to take place in the localized zones through which water penetrates. While this localized calcium leaching does not affect the overall structural performance of the slab, it may contribute to further weakening the construction joints, and may eventually have degraded the grout in the annular space between the 3 in diameter hole and the 2 in diameter Williams rock anchors. A close-up inspection of the grout and bolt is recommended.

4. The lower pH-value of 9.3–9.4 of the water emerging from localized zones along the construction joints, compared to the typical pH~12 of concrete's bulk pore solution, is consistent with the calcium leaching observation. Its localized occurrence does not compromise the corrosion protection of the steel reinforcement in the slab. A refined corrosion indicator analysis is recommended to confirm the prevention or minimization of reinforcement and anchor bolt corrosion.

5. Changes in environmental conditions (e.g., seasonal changes in water table or a seismic event) that affect the static head that drives the water migration through the concrete would impact the rate of water seepage into the torus room. These affects would be small since, as discussed in the report, the discontinuities in the concrete base mat that are allowing the water seepage into the torus room are very small. Even if the current very low rate of water intrusion increased by an order of magnitude because of a change in static head there would be no impact on plant safety due to the large size of the torus room.

1. Objective

The objective of this report is a preliminary durability evaluation of the Torus Mat in Entergy's Pilgrim Station. This report is based on observations and documents shared by the Entergy team with the author during and following the visit on Monday, March 19, 2007 to Entergy's offices in the Pilgrim Station.

2. Observations

The 8 ft thick Torus mat is part of the reactor building foundation, and surrounds the much thicker core vessel foundation. The Torus mat is divided into 16 Torus room bays, with bay #2 being the most south, and bay #10 the most north, where the ocean is situated.

At any point in time water may be observed on the floor of one or more Torus room bays, puddled at the low point under the invert of the Torus. The maximum water depth may be about ½ inch or more, but generally limited to the area directly beneath the Torus shell. The wetted areas of the floor have not extended out from under the Torus, and pumping for removal has never been required. Some of remaining bays having no water may have crystal residue indicating they were once wet. Several other bays show no evidence to suggest that they were ever wet. These are bays #16, 1, 2, 3 and 4, which are all situated on the south side of the reactor foundation, i.e. the furthest away from the sea.

Conditions associated with water seepage into the Torus room have been reported in the early 1980s but may have existed even before then. In the early 1980s seepage was reported at the junction of the drywell pedestal and the Torus room floor, in Bay #15. In the late 1990s, Torus room conditions were evaluated by Engineering. The most probable cause of water seepage was attributed to groundwater by-passing the waterproof membrane that encapsulates the Reactor Building 8 ft. thick base mat. A 25 ft. hydrostatic head forces water through the mat at discontinuities such as construction joints, anchor bolts holes or other features that result in reduced head losses. This conclusion was based on the following observations and operating experience with membrane designs similar to Pilgrim:

- A portion of the floor in Bay #10, adjacent to the Torus saddle common to Bay #11 (situated on the north side), was dried, cleaned and isolated with a berm. Within the berm is a Williams rock anchor installed in the early 1980s to secure the Torus saddles for pipe break accident uplift conditions. Within about a day, water reappeared on the floor within the berm. There were no visible cracks on the floor surface, suggesting that the water was coming up from the anchor hole drilled in the slab, or from beneath the torus saddle base plate.
- The water was sampled and not found to be contaminated to an extent that would indicate an active plant system leak. Some radioactive contamination was found but this was believed to be the result of plant system leaks that occurred much earlier in the plants operating history.
- There were no active plant system leaks that could result in water on the Torus room floor.
- The seepage rate appears to be at steady state with evaporative losses, hence the floor surface within any bay never fills completely.

3. Methodology

Based on these observation, and additional information about the construction history of the reactor building foundation, the tasks of this preliminary durability performance evaluation are:

(1) To identify the most likely cause of the observed seepage into some Torus room bays, and

(2) To estimate the effect of this cause on the integrity and performance of the base mat foundation.

(3) Finally, some preliminary conclusions are drawn together with recommendations for further investigations.

4. Analysis

4.1 Correlating Construction History and Occurrence of Water Seepage

The job drawings of the construction sequence and photos taken during the construction indicate that the Torus mat was constructed first before the massive core foundation was put in place. In particular, the construction of the Torus mat was divided in four concrete segments, staring with the North-West Corner, followed by the South-East Corner, the North-East Corner and finally the South-West Corner. Finally, the central core foundation was poured. As such there are a total number of four construction joints between the four mat segments, in addition to the joint between the core foundation and the surrounding Torus base mat. It has been earlier suggested that those construction joints could be the preferential path for water seepage.

In order to check this suggestion, Figure 1 overlays in a plan view the pouring sequence with concrete joints and the bays in which water seepage has been observed. There appears to be a clear correlation between the construction sequence and the locus of occurrence of water. In particular:

(1) The water seepage observed in Bay #6 and #10, situated respectively on the North and the East side of the reactor foundation are situated in bays delineated to the inside by the core-mat construction joint, and separated in the center of the bay by one mat-to-mat construction segment joint.

(2) The water seepage observed in bays #7, #11 and #13 are all delineated to the center by a core-mat construction joint characterized by a kink forming a more-or-less sharp wedged geometrical discontinuity.

There appears to be, therefore, a strong indication in favor of the suggestion that the preferential paths for water seepage are construction joints. From a concrete material perspective, concrete construction joints are well-known to be the weak spot of any concrete engineering application, due to so-called "wall effect", leading during pouring of the fresh concrete to a higher water-concentration compared to bulk concrete. This higher water concentration available for cement hydration leads to a higher final porosity of the concrete in the immediate surrounding (typically a fraction of 1in) of the joints than in the concrete bulk. Since the flow of water occurs through this porosity, concrete construction joints form a preferential path for fluid conduction, as analyzed later on. With regard to the occurrence of water in specific bay areas, the following additional observations are made:

(1) In bay #6 and #10, it is most likely that the water seepage originates from the 'T' junction of the mat-mat joint with the core-mat joint.

(2) The fact that the core foundation was constructed only after the mat may have had some important implications on the stresses that were most likely generated during the concrete hydration. In fact, since hydration is an exothermic reaction, heat is generated during hydration, which leads to high temperature rises in massive concrete elements. This heat is transported through the concrete surface leading to a cooling over time of the concrete, until the concrete element reaches the ambient temperature [1]. Given the construction process, the base mat which had been poured first, must have been already in a state of cooling, when the much thicker core foundation was poured. Given the massive dimensions of the core foundation, the temperature in the core foundation is expected to have followed an almost adiabatic temperature rise. This temperature differential between the cooler mat and the hot core generates circumferential tension stresses in the mat immediately adjacent to the concrete joint. Indeed, a rough analysis of the stresses surrounding the core foundation shows that the circumferential (or hoop) stresses in the immediate vicinity of the joint are in tension if the temperature rise in the core had been twice the temperature rise in the mat. These stresses are amplified, in bay #7,#11,#13 by the presence of the wedge-shaped joint, leading to a significant stress amplification due to stress concentration which is typical for geometrical discontinuities. It is, therefore most likely that those wedge-shaped corners may have been additionally weakened during the differential hydration, forming some preferential path for water seepage.

(3) The fact that most water seepage observed occurred on the North side may well be related to the direction of water flow below the foundation, flowing towards the sea situated on the North side.

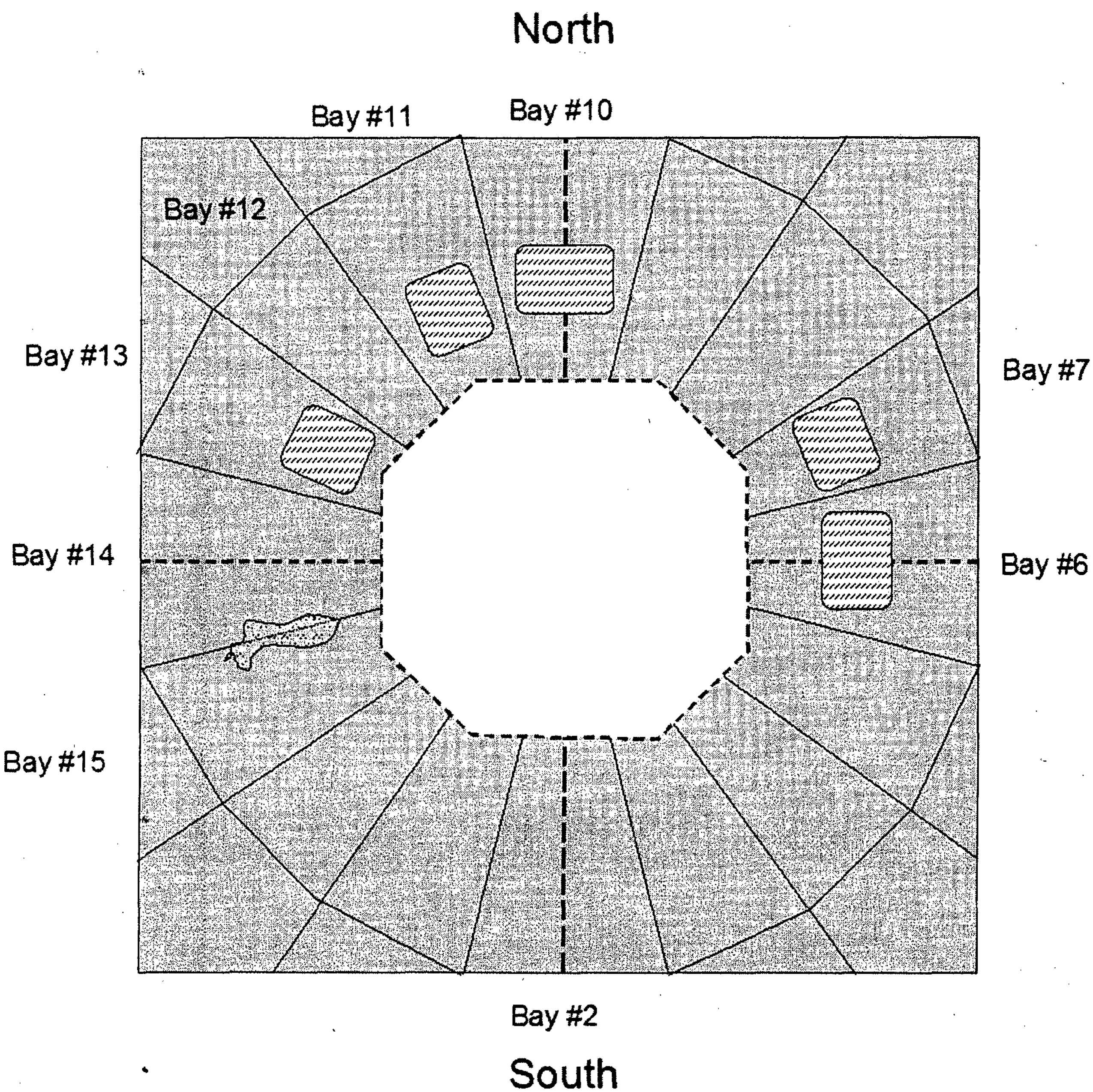

Figure 1: Correlating concrete construction sequence with water seepage occurrence: The dashed lines represent construction joints (re-constructed from shop-drawings), the patches represent observed water seepage in specific bays.

It should be noted that those phenomena (higher porosity of concrete close to joints, microcracks around geometrical discontinuities) are almost inevitable in massive concrete engineering applications due to the high exothermic nature of cement hydration. It is for this reason that concrete design codes specify a minimum amount of steel reinforcement, which needs to bridge concrete construction joints. This reinforcement ensures the structural performance of the slab, despite some localized concrete material weakness along construction joints. This reinforcement, however, cannot eliminate neither the higher material porosity, nor the occurrence of microcracks. It may eventually limit the microcrack opening and propagation; thus ensuring the structural performance of the concrete foundation.

4.2 Seepage through Concrete Bulk and Construction Joints

The most likely source of the water seepage observed on the Torus floor is due to groundwater infiltration driven by the hydraulic head of the pore fluid under gravity forces. For purpose of analysis, the following assumptions are made:

- The porosity of the 8 ft concrete mat is assumed to be fully saturated by liquid water. This seems to be a reasonable assumption, given that the 8 ft thick concrete slab would take some centuries1 to dry to a level in equilibrium with the ambient humidity conditions inside the Torus room (see, for instance, [2,3]).

- The 8 ft concrete slab is sufficiently homogeneous throughout its thickness. This allows one to condense the flow into a single material parameter, the intrinsic permeability of concrete. For bulk concrete, typically values for the intrinsic permeability reported in the open literature vary between 10^{-17}–10^{-16} m^2, depending on the concrete mix proportion and curing conditions (see, for instance, [4]). The concrete composition and massiveness of the Torus base mat is indicative of an intrinsic permeability of 10^{-17} m^2.

Under these assumptions the mass flux rate through the 8 ft concrete slab can be estimated using Darcy's Law. For reference, we first estimate the mass flux rate through the concrete bulk. For an hydraulic head difference of 24.5 ft between the bottom of the slab and the surface and an intrinsic permeability of 10^{-17} m^2, these calculations yield a mass flux (per unit surface) through the concrete bulk of Q=0.02 kg/ (m^2day) which amounts to 6.2 kg/ (m^2yr). Such a small amount of water (which generates daily a water film 17μm thin) is expected to evaporate almost instantaneously, and it is a clear indication that the water flux through the concrete bulk porosity cannot be at the origin of the observed water seepage. It hints towards a very localized nature of the water penetration. Furthermore, the small value is a benchmark value for a first-order estimate of the size of the joint openings through which the water flow occurs.

Indeed, it was observed that in a previously dried area water reappeared within a day or two generating a water film on-average 1/4 in thick2. This observation translates into a water flux of Q=3.2-6.4 kg/(m^2day), which is substantially greater than the water flux through the concrete bulk.

One can attempt to link this high flux rate to the space through which the flow occurs, by considering the flow of water through a cylinder of radius a in a cross section A (see, for instance [5]). In this simple model representation of the flow along the joints, the cylinder represents the joint opening, while the cross section represents the wetted surface in consequence of the flow through the cylinder. These calculations provide an upper-bound estimate of the cylinder radius representing the characteristic size of the joint opening, and yield values for the pore throat radius in functions of the wetted surface area. The results which are displayed in Figure 2, indicate that the equivalent cylindrical "joint" opening, is on the order of 1-2 mm (~1/25 – 1/12 in) for wetted surface areas of 5-20 m^2 (~50–200 ft^2) which was observed in some bay areas.

It should be noted that this rough model overestimates the joint opening because it cumulates all possible joint openings into one single cylindrical shape. On the other hand, it provides an upper-bound estimate of the order of magnitude of the joint opening, which is expected to be in the sub-millimeter range, but substantially greater than the typical capillary pore size of concrete which is in the micrometer range.

In all cases, the order of magnitude estimations of the water flow generated by the hydraulic head difference is clear evidence of a mal-functioning of the water-stop PVC membrane originally designed to control seepage at concrete construction joints.

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1 Drying of concrete is an extremely slow process. It takes some 10 years to dry a slab of 12 cm thickness exposed on both sides to ambient conditions. The drying duration increases with the square of the thickness. Hence, over the last 35 years, the drying front in the torus room may have reached a depth of $x = (35/10)^{0.5} \times 6 = 11.2$ cm = 4.4 in, which is negligible (5%) compared to the base mat thickness of 8 ft. In return, this drying may have caused some microcracking (almost invisible to the eye) in the concrete surface, expanding roughly half the depth of the drying front [3]. This microcracking scales with the mass loss and is little affected by an increase of the amount of steel reinforcement.

2 John Dyckman, email March 16, 2007.

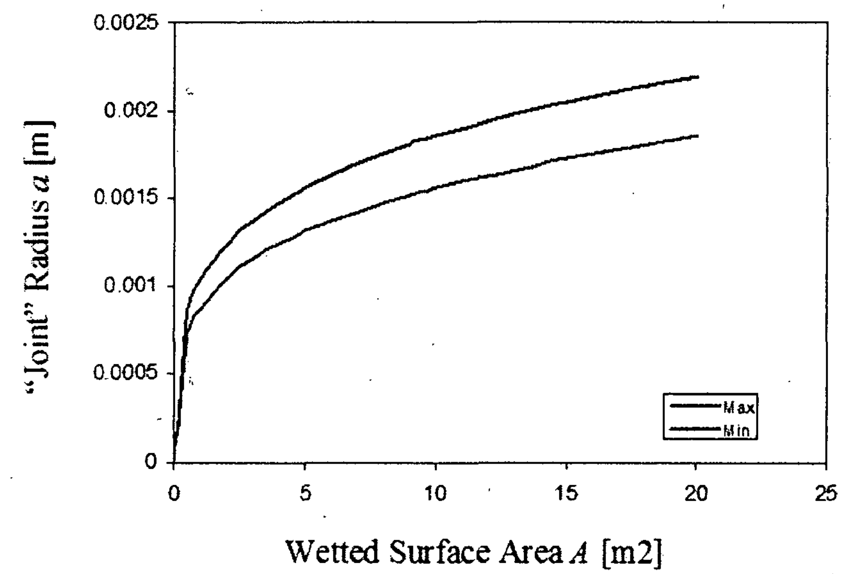

Figure 2: Estimated equivalent cylindrical joint size as a function of the wetted surface area. The upper curve is based on water re-appearance within one day, the lower curve on water re-appearance in two days.

4.3 Effect of Calcium Leaching

When concrete is put in contact with water having a lower calcium concentration than the equilibrium calcium concentration, calcium is leached from the concrete into the pore solution. The equilibrium concentration of calcium in the pore solution is roughly 480 mg/L (see e.g. [6-7]), meaning that if the calcium concentration in the pore solution is below this threshold value calcium is dissolved ("leached") from the solid into the pore solution. The consequence of leaching is a substantial increase in the porosity (due to the dissolution of Portlandite) and a substantial loss in mechanical stiffness and strength properties [6,8].

Water collected from the Torus room and analyzed chemically[3] showed a calcium concentration of 230 mg/L, which is smaller than the equilibrium calcium concentration. As a consequence, it cannot be excluded that calcium leaching occurred along the preferential path of groundwater intrusion. In favor of this suggestion is the observation that evaporation residues furthermore show a 31mw% of calcium[4].

Fortunately, calcium leaching is a very slow process: the calcium leaching front advances 0.115 mm/√day; which means that over the last 35 years the leaching may have dissolved the calcium in a layer maximum 13 mm (~1/2 in) thick. In other words, in the life span of the power plant, calcium leaching has no effect on the bulk integrity and structural performance of the concrete foundation. On the other hand, it cannot be excluded that calcium leaching may have contributed to the weakening of the construction joints over the years. In fact, the calcium leaching may have contributed to the joint opening through which groundwater (at a lower calcium concentration than the equilibrium concentration) penetrates. The way by which calcium leaching is most likely to affect the joint opening is sketched in Figure 3, showing dissolution fronts originating from the bottom side around a joint (left). The figure shows that the dissolution generates a vertical wedge-shaped dissolution pattern around the joint, which propagates upward through the slab in a self-similar fashion. These dissolution fronts scale linearly with the water-velocity in the joint and the joint opening [9]; which means that the higher the water flow through the joint, the more advanced the vertical position of the degraded zone.

Measurements of the flow rate through the joints could make it possible to estimate the current height of the leaching front in the 8 ft base mat. In the absence of such measurements, it is not possible to exclude that leaching by groundwater may have reached the grout of the Williams rock anchored (2ft below surface), leaching the calcium of the grout in the annular space between the 3 in diameter hole and the 2 in Williams rock anchor.

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3 Northeast laboratory Services Report, dated 03/19/2007.

4 Northeast Laboratory Services Report, dated 03/15/2007.

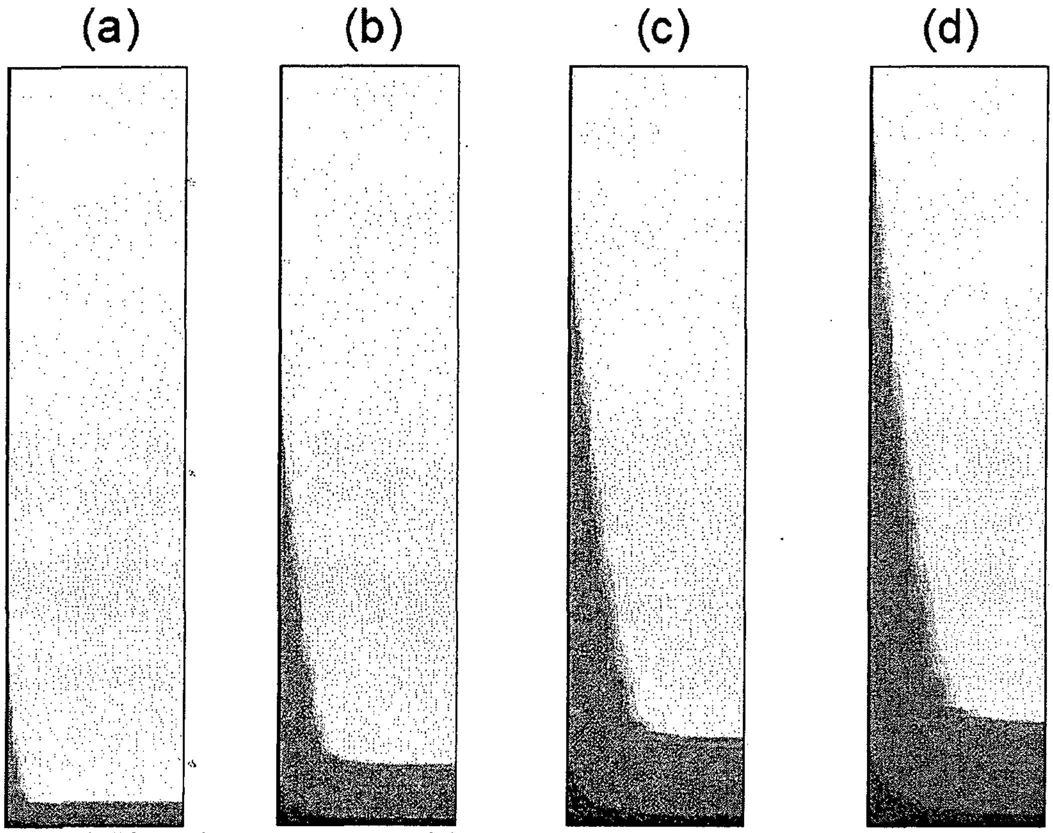

Figure 3: Dissolution Fronts around a joint through which water flows with a lower calcium concentration than the equilibrium concentration. For a flow velocity of V=10 cm/d and a joint opening of 0.4 mm, (a) 27 years; (b) 158 years; (c) 318 years; (d) 438 years. The vertical height in this figure is 5 cm (reproduced from [9]).

4.4 Effect of pH Value

Compared to the groundwater (pH ~ 6.7)5, the water collected in the Torus bay has a pH value of 9.3 – 9.46. This pH value is below the typical pH~12 value found in the highly basic interstitial pore solution of concrete, and it is consistent with the observation of a lower-than-equilibrium calcium concentration in the water seepage. Given the very low flow rates through the bulk of the concrete slab, it is reasonable to expect that a high pH~12 value prevails in the pore solution of the concrete bulk, while the slightly lower pH value of 9.3 – 9.4 only occurs locally in the water penetrating through the joint openings. In all cases, both pH values are in a range that should prevent or minimize reinforcement corrosion. To ascertain this suggestion, a refined chemical analysis of the water may be helpful, one in which not only the pH value is measured (which represents the H^+ cation concentration), but as well other quantities, such as the Cl^-/OH^- concentration ratio in water. Such corrosion indicators have recently been identified to provide a more comprehensive measure of the onset of corrosion, as it requires a critical amount of Cl^- to start the corrosion (see, for instance, [10] and references cited herein).

5. Summary of Analysis and Recommendation

The analysis of observations and documents relating to the groundwater intrusion into some bays of the Torus room allows for the following preliminary conclusions:

1. The observed groundwater penetration is a highly localized phenomenon. It is caused by the high hydraulic head difference, pushing groundwater through vertical joints and zones most likely weakened by tensions

5 SAIC Report dated July 17, 2006.
6 Northeast laboratory Services Report, dated 03/19/2007

generated during setting and hydration following the construction. These zones are discontinuities equivalent to a vertical cylindrical hole of a maximum diameter of 4 mm (1/6th in).

2. The calcium concentration of the collected groundwater is smaller than the equilibrium calcium concentration in cementitious materials. As a consequence calcium leaching is expected to take place. The highly localized nature of the water penetration ensures that this leaching has no effect neither on the bulk integrity of the concrete, nor on the overall structural performance of the reactor foundation. (In fact, a vertical cylinder of 4 mm diameter in a ~142 ft x 142 ft foundation slab, will not compromise neither its compressive load distribution function nor the bending capacity of the slab.)

3. Thus, the effects of calcium leaching are localized around the vertical joints and other vertical discontinuities present in the slab. Depending on the flow velocity, leaching may have affected the grout in the annular space between the 3 in diameter hole and the 2 in diameter Williams rock anchor, compromising the grout's stiffness and strength. It is recommended to inspect whether the grout has been chemically degraded.

4. The lower pH-value of the collected groundwater is consistent with the observation of a lower-than-equilibrium calcium concentration. Since the phenomenon is localized around some weak spots along the construction joints, it is unlikely that this locally lower pH value substantially increases the risk of reinforcement corrosion. To further prevent and minimize the risk of reinforcement corrosion, a refined chemical analysis is recommended to measure relevant corrosion indicators, such as the Cl^-/OH^- concentration ratio, and to compare those corrosion indicators with now well-established corrosion thresholds [10]. A close-up inspection of the Williams rock anchors may complement this evidence of a non-detectable corrosion risk.

In summary, the analysis provides evidence that the observed groundwater penetration does not compromise the structural performance of the Torus base mat. It is recommended that any corrective measurement should be based on a prior identification of the exact location of the weak points along the construction joints, and a detailed analysis of the degradation state of both the grout and the bolt of the William rock anchor system.

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ATTACHMENT E to Letter 2.07.027

(22 pages)

Structural Integrity Associates Fluence Evaluation for PNPS

Structural Integrity Associates, Inc.	File No.: PNPS-27Q-301
CALCULATION PACKAGE	**Project No.: PNPS-27Q**

PROJECT NAME: Evaluation of Fluence Issues for Pilgrim Nuclear Power Station

CONTRACT NO.: 4500555337, Rev. 1

CLIENT: Entergy Nuclear Northeast	**PLANT: Pilgrim Nuclear Power Station**

CALCULATION TITLE: Evaluation of Fluence Issues for PNPS

Document Revision	Affected Pages	Revision Description	Project Manager Approval Signature & Date	Preparer(s) & Checker(s) Signatures & Date
1	3, 8, 9, 10, & 12	Update Sections 1.0, 4.5, 5.0, 6.0, and 7.0	T. J. Griesbach 4/30/07 Timothy J. Griesbach	T. J. Griesbach 4/30/07 Timothy J. Griesbach G. L. Stevens 4/30/07 Gary L. Stevens

Contains References to Proprietary Information

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1.0 INTRODUCTION

The recent fluence re-evaluation for the Pilgrim Nuclear Power Station (PNPS) reactor pressure vessel (RPV) using the EPRI RAMA code required an increased fluence bias correction factor (CF) of 1.78 to adjust for the benchmarking discrepancy with the cycle 4 surveillance capsule dosimetry results [1]. A rigorous technical explanation for this bias has not been determined. As a result, the NRC will not accept the PNPS fluence calculations for application to future plant operation without further justification. In response to this, Entergy Nuclear Northeast (ENN) has requested SI to perform an additional evaluation to demonstrate adequate vessel life prediction through the extended 60-year operating period with respect to the fluence projections.

Increasing fluence has an effect on the toughness of the RPV materials. This is measured by an increase in the adjusted reference temperature (ART) and a decrease in the upper shelf energy (USE) of the RPV beltline materials. The PNPS FSAR identifies the vessel as being controlling for all reactor pressure boundary carbon steel components [17]. The ASME Code [2] and 10CFR50, Appendix G [3] give criteria for maintaining pressure boundary integrity including the effects of materials degradation due to irradiation damage. Additional evaluations for equivalent margins have been submitted to the Nuclear Regulatory Commission (NRC) and approved for use by boiling water reactors (BWRs) for Charpy USE drop. These equivalent margin analyses for USE are published in BWRVIP-74-A [4]. In addition, BWRVIP-05 [5] provides a technical basis for alternative inspection requirements of the RPV shell welds to eliminate inspections of circumferential welds. The methods and criteria in these documents form the basis for demonstrating vessel integrity margins, including the effects of plant aging due to fluence.

ENN performed an integrated plant assessment (IPE) to extend the operating license of PNPS. This included a review of the time-limited aging analyses (TLAA) and exemptions to 10CFR50 for the period of extended operation [6]. Increasing fluence is one aspect considered in the TLAAs. The calculated fluence in the vessel using the method from Reference [1] is projected through 54 effective full power years (EFPY) without a bias correction factor. The results of that study are now being reevaluated using assumed fluences greater than the previously calculated results. This analysis for PNPS uses the established methods and criteria for evaluating embrittlement for fluence levels exceeding the previously projected end-of-license fluence in the vessel.

2.0 TECHNICAL APPROACH

The shift in the ART and a decrease in the USE for ferritic materials are predicted by Regulatory Guide 1.99, Revision 2 [7]. The embrittlement trend curves are a function of copper (Cu) content, nickel (Ni) content, and fluence; different trend curves apply for welds and base metals. The materials in the RPV that must be monitored for irradiation effects are the regions where significant fluence levels are projected (> $1x10^{17}$ n/cm^2, E > 1 MeV), and those materials are characterized as beltline materials. Analyses of all of the beltline materials for the PNPS vessel determine the weld or plate that is the limiting RPV beltline material. The properties of that limiting beltline material are then used to calculate the operating heatup, cooldown, and pressure test curves. Those calculations were performed previously for the PNPS RPV for a fluence up to 54 EFPY using fluence projections with and without the 1.78 bias correction factor [8, 9]. The calculations show that there is no RPV integrity concern for 54 EFPY even with the bias corrected fluence.

To further demonstrate that the fluence uncertainty issue for PNPS is not a concern, additional analyses are being performed in this calculation assuming even greater fluence levels in the RPV beyond the 54 EFPY predicted fluence values with the 1.78 bias correction factor. The fluence levels are assumed to increase until a criterion for operability can no longer be maintained. When that limit is determined, the calculated factor on fluence is an indication of the conservatism against brittle fracture of the RPV (or some other criteria) in order to accommodate the observed uncertainty in the fluence calculations.

3.0 ASSUMPTIONS / DESIGN INPUTS

1. The pressure for the pressure test is normal operating pressure (1,035 psig) from Reference [10].
2. The maximum test temperature for the hydrotest is 212°F per the PNPS Technical Specifications [11]. (Note that this is an operational limit, not a brittle fracture limit.)

4.0 CALCULATIONS

4.1 Maximum Fluence to Perform Hydrotest

Irradiation by high energy neutrons raises the RT_{NDT} of the reactor vessel materials. The ART is defined as $RT_{NDT} + \Delta RT_{NDT}$ + Margin in accordance with Regulatory Guide 1.99, Rev. 2 [7]. The pressure-temperature (P-T) curves are developed from the ART value for the vessel material. The calculated hydrotest pressure vs. temperature curve (Curve A) results for 54 EFPY are shown in Table 1 and in Figure 1 [8]. The PNPS projected values for ΔRT_{NDT} and ART at 54 EFPY were calculated with the 1.78 bias correction factor on fluence [9]. The projected values of ART are shown in Table 2. The hydrotest pressure is the normal operating pressure, which is 1,035 psig [10]. The system hydrostatic test temperature is calculated to meet the requirements of ASME Section XI, Appendix G, Article G-2400 [2]. The system hydrostatic test should be performed at a temperature not lower than the highest required temperature for any component in the system. For PNPS, the limiting component is the beltline material with the highest ART value at the quarter-thickness (1/4t) location. From Table 2, the limiting materials are the lower intermediate shell longitudinal welds #1 and #3.

The maximum calculated ART value for these welds at 54 EFPY is 122.7°F. This corresponds to a 1/4t fluence value of 1.46×10^{18} n/cm^2, including the 1.78 bias correction factor. The hydrotest temperature at this fluence is 152.5°F. This hydrotest temperature is interpolated linearly from the values from Table 1 as follows:

Hydrotest Temperature (°F)	Hydrotest Pressure (psig)
150	1,007
152.5	**1,035**
155	1,063

The temperature difference between the 1/4t ART value and the hydrotest temperature is calculated to be 29.8°F. This temperature difference is assumed to be constant for increasing fluence and ART values, so the maximum fluence to conduct the hydrotest can be calculated from the maximum achievable temperature to perform the hydrotest, which is 212°F for PNPS [11]. The 1/4t fluence and corresponding 1/4t ART for the limiting welds are increased until the hydrotest temperature of 212°F is reached. From the table below it is noted that the maximum 1/4t fluence of 4.12×10^{18} n/cm^2 corresponds to a 1/4t ART value of 182.2°F for a hydrotest temperature of 212°F, the maximum temperature to perform the hydrotest at PNPS.

Calculation of Hydrotest Maximum Temperature and Fluence

1/4t Fluence (n/cm^2)	1/4t ART (°F)	Hydrotest Temp. (°F)	Temp. Difference (°F)	Fluence Ratio	
1.46E+18	122.7	152.5	29.8	1.00	
2.00E+18	139.6	169.4	29.8	1.37	
3.00E+18	162.9	192.7	29.8	2.05	
4.00E+18	180.4	210.2	29.8	2.74	
4.12E+18	**182.2**	**212**	**29.8**	**2.82**	maximum fluence to conduct hydrotest < 212°F
4.50E+18	187.8	217.6	29.8	3.08	
5.00E+18	194.4	224.2	29.8	3.42	

The calculated hydrotest temperature and 1/4t ART values versus fluence are shown in Figure 2. A fluence ratio of 2.82 is the ratio of the maximum 1/4t fluence at the limiting vessel beltline welds compared to the 54 EFPY fluence with the 1.78 bias correction factor. In other words, the fluence with the 1.78 bias correction factor would have to be increased by an additional factor of 2.82 before the limiting hydrotest temperature of 212°F is reached.

4.2 Maximum Fluence to Maintain Charpy Upper Shelf Energy

Appendix G of 10CFR50 requires that reactor vessel beltline materials "have Charpy upper shelf energy...of no less than 75 ft-lb initially and must maintain Charpy upper shelf energy throughout the life of the vessel of no less than 50 ft-lb." Regulatory Guide 1.99, Rev. 2, *Radiation Embrittlement of Reactor Vessel Materials,* defines the method for predicting upper shelf energy drop in terms of a percentage from the unirradiated value. Figure 3 shows the predicted Charpy upper shelf energy for welds and base metals as a function of copper content and fluence.

The predicted Charpy upper shelf energy (C_VUSE) values for PNPS at 54 EFPY were determined previously for the PNPS license renewal project [6]. The predicted C_VUSE values based on the Regulatory Guide 1.99 Position 1 method are shown in Table 3. The predicted values for C_VUSE using the 54 EFPY fluences with the 1.78 bias correction factor are shown in Table 4. It is noted that all projected USE values are above 50 ft-lbs, even with the 1.78 bias correction factor on fluence. The USE limit shows a minimum fluence ratio of 4.9 for the projected fluence to reach 50 ft-lbs for the lower intermediate shell axial welds, as shown in Table 5. Because the USE values are always greater than 50 ft-lbs., the equivalent margin method of BWRVIP-74-A is not required.

4.3 Maximum Fluence Bounded by the Reactor Vessel Weld Failure Probability

The BWVIP recommendations for inspection of reactor vessel shell welds in BWRVIP-05 [5] are based on generic analyses supporting a Safety Evaluation Review (SER) conclusion that the generic plant axial weld failure rate is no greater than $5x10^{-6}$ per reactor year [12] at the end of 40 years. BWRVIP-05 showed that this axial weld failure rate is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above.

PNPS received relief from the circumferential weld inspections for the remainder of the original 40-year operating term [13]. The basis for this relief request was a plant specific analysis that showed the limiting conditional failure probability for the PNPS circumferential welds at the end of the original operating term were less than the values calculated in the BWRVIP-05 SER [12].

Table 6 contains a comparison of the PNPS reactor vessel limiting axial weld parameters to those used in the NRC analysis. The data in column two (CE) is from Table 2.6-5 of the NRC SER for BWRVIP-05 [12]. The data in the third column (PNPS) is the projected 54 EFPY data for PNPS without the 1.78 bias correction factor on fluence [6]. (For consistency with the NRC evaluation, the RT_{NDT} is calculated without the margin term.) The data in column four (PNPS with Bias CF) is the projected 54 EFPY data for PNPS with the 1.78 bias correction factor on fluence.

Column five (PNPS Limit) shows the maximum fluence and RT_{NDT} to assure that the limiting axial weld remains bounded by the value of 172.4°F determined from the CEOG and accepted in the SER [12]. The maximum ID fluence of 8.48×10^{18} n/cm^2 gives a fluence ratio of 4.18 compared to the 54 EFPY fluence with the 1.78 bias correction factor.

Table 7 contains a comparison of the PNPS reactor vessel limiting circumferential weld parameters to those used in the NRC analysis. The data in column two (CE) is from Table 2.6-5 of the NRC SER for BWRVIP-05 [12]. The data in the third column (PNPS) is the projected 54 EFPY data for the PNPS circumferential weld without the 1.78 bias correction factor on fluence [6]. The data in column four (PNPS with Bias CF) is the projected 54 EFPY data for the PNPS circumferential weld with the 1.78 bias correction factor on fluence. Column five (PNPS Limit) shows the maximum fluence and RT_{NDT} to assure that the PNPS circumferential weld remains bounded by the value of 128.5°F determined from the CEOG and accepted in the SER [12]. The maximum ID fluence of 1.14×10^{19} n/cm^2 gives a fluence ratio of 7.35 compared to the 54 EFPY fluence with the 1.78 bias correction factor.

PNPS obtained relief from the examination of RPV circumferential welds related to the augmented shell weld examination requirements contained in 10CFR50.55a(g)(6)(ii)(A)(5). The reduction in scope of these inspections from essentially 100 percent of all RPV shell welds to examination of essentially 100 percent of the axial welds and essentially zero percent of the circumferential welds was based on the NRC staff determination that the conditional probability of failure for these welds was within the acceptable limits at the expiration of the current operating license [13]. The results given in Tables 6 and 7 show that the bounding reactor vessel weld conditional failure probabilities can be maintained well beyond the 54 EFPY projected fluences and ART values for the PNPS vessel. The relatively large calculated fluence ratios shown in these tables indicate that the criteria for relief from the circumferential vessel weld inspections will not be the limiting factor for fluence margin in the PNPS RPV.

4.4 Effect of Fluence on Evaluation of N2 Nozzles

The fluence levels in the N2 nozzles are relatively low compared to the peak fluence in the beltline. These fluences shown in the table below were obtained from the RAMA code fluence calculation [9, 14].

	54 EFPY Fluence @ 1/4t (w/o 1.78 bias CF) (n/cm^2)	54 EFPY Fluence @ 1/4t (with 1.78 bias CF) (n/cm^2)
Recirc. inlet (N2) nozzles	2.02E+17	3.60E+17
Limiting Axial Welds	8.18E+17	1.46E+18

The effect of the increasing fluence on the calculated ART values for the limiting weld and the N2 nozzles is shown below. The ART values for the A508-2 nozzle forgings was estimated using upper bound Cu = 0.35, Ni = 0.85, and an initial RT_{NDT} of 0°F [14].

	54 EFPY ART @ 1/4t (w/o 1.78 bias CF) (°F)	54 EFPY ART @ 1/4t (with 1.78 bias CF) (°F)
Recirc. inlet (N2) nozzles	77.0	94.7
Limiting Axial Welds	95.3	122.7

Structural Integrity Associates recently performed an evaluation of the recirc. inlet nozzles using best estimate copper and nickel chemistry values of Cu = 0.15 wt%, Ni = 0.85 wt% [14]. Using these best estimate values, the calculated ART values for the nozzles are as follows:

	54 EFPY ART @ 1/4t (w/o 1.78 bias CF) (°F)	54 EFPY ART @ 1/4t (with 1.78 bias CF) (°F)
Recirc. inlet (N2) nozzles	39.9	56.4
Limiting Axial Welds	95.3	122.7

From the comparison of the ART values for the recirc. inlet nozzles and the limiting axial welds, the recirc. inlet nozzle embrittlement levels are well below the projected ART values for the limiting axial welds. This is mainly because of significantly lower fluences at the height of the nozzles compared to the active core region. Thus, there is no impact of fluence uncertainty for this evaluation, and it is determined that the nozzles will not become the limiting beltline materials for P-T limits or hydrotest conditions as fluence levels are increased.

4.5 Effect of Fluence on RPV Internals

4.5.1 Top Guide

BWRVIP-26 calculated the minimum top guide fluence for 32 EFPY (40 years) as 4×10^{21} n/cm^2 [15]. The threshold for IASCC is 5×10^{20} n/cm^2, and the PNPS top guide fluence will exceed this threshold [6]. Therefore, PNPS must manage IASCC of the top guide assembly. PNPS has implemented the inspection recommendation in BWRVIP-26 through the BWR Vessel Internals Program [16]. The BWR Vessel Internals Program will adequately manage the effects of aging on the top guide for the period of extended operation. The top guide does not affect the operating P-T limit curves, and there is no criterion on fluence that would further limit the operation of the top guide structure.

4.5.2 Core Shroud

The core shroud is a BWR component that is known to be susceptible to aging effects. Section 3.8.12 of the PNPS License Renewal Project, TLAA and Exemption Evaluations [6] addresses the time limited aging analyses of the core shroud. A review of the analyses related to the core shroud found that the only TLAA involves the fatigue analysis and calculation of cumulative usage factors (CUFs) for the shroud repair. The core shroud does not affect the operating P-T limit curves, and there is no criterion on fluence that would further limit the operation of the core shroud structure.

5.0 RESULTS OF ANALYSIS

The effects of increased fluence beyond the projected 54 EFPY fluence calculations for the PNPS RPV are summarized below for each of the potential aging effects. The results are compared to determine the minimum acceptable fluence ratio. This is the fluence multiplier that could be achieved compared to the 54 EFPY fluence with the 1.78 bias correction factor, and is the measure of tolerance on fluence before a limit is reached that would exceed a Code limit, regulatory criterion, or service limit.

Effect of Fluence on	Acceptable Fluence Ratio
Hydrotest Temperature	2.82*
Charpy Upper Shelf Energy	4.86
RPV Axial Weld Failure Probability	4.18
RPV Circ. Weld Failure Probability	7.35
Evaluation of N2 Nozzles	Bounded by beltline

* minimum acceptable fluence ratio = 2.82

6.0 CONCLUSIONS AND DISCUSSIONS

Fluence contributes to changes in the vessel beltline material properties. These changes are measured by the shift in RT_{NDT} or the drop in USE of the ferritic materials (i.e., welds, plates, and forgings). The analyses using projected fluence values for license renewal (54 EFPY) for PNPS show no limitations due to embrittlement concerns for the vessel. Considering increasing fluence levels, the RPV analyses demonstrate that the Code and regulatory criteria can be met for operation well beyond this maximum fluence level by a factor of 2.82 (or greater) on the 54 EFPY fluence including a bias correction factor of 1.78.

The limiting condition for the vessel is the temperature required to perform the ASME Code hydrotest. The temperature to perform the hydrotest is prescribed by ASME Section XI, Article G-2400 that requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The maximum temperature limit for the hydrotest of 212°F in the PNPS Technical Specifications is an administrative limit; it may be possible to perform the test at higher temperatures which would allow for even higher fluence levels.

These analyses demonstrate that there is a considerable tolerance on the acceptable range of fluence. This is exemplified by the difference between the fluence for the maximum predicted levels of embrittlement and the limiting criteria for operability, a difference large enough to accommodate the uncertainties on the calculated fluence for PNPS.

7.0 REFERENCES

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2. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 1998 Edition, 2000 Addenda.

3. Code of Federal Regulations, 10 CFR, Part 50, Appendix G, Federal Register: January 2005.

4. BWRVIP-74-A, "BWR Reactor Vessel Inspection and Flaw Evaluation Guidelines for License Renewal, BWR Vessel and Internals Project," **EPRI Proprietary**, 1008872, June 2003, (SI File No. BWRVIP-01-274AP).

5. BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," BWR Vessel and Internals Project," **EPRI Proprietary**, EPRI TR-105697, September 1995. (SI File No. BWRVIP-01-205P).

6. PNPS License Renewal Project, TLAA and Exemption Evaluations, LRPD-03, Revision 0, January 2006, (SI File No. PNPS-27Q-202).

7. NRC Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

8. SI Calculation PNPS-03Q-301, Revision 1, "Development of Pressure Test (Curve A) P-T Curves," 1/30/06.

9. SI Calculation PNPS-22Q-301, "ΔRT_{NDT} and ART Evaluation," 1/20/06.

10. Email from Bryan Ford to Raymond Pace, Timothy J. Griesbach, and Gary L. Stevens, Subject: Maximum Pressure Test Temperature, dated 3/8/07, (SI File No. PNPS-27Q-205).

11. PNPS Technical Specifications, Revision 274, Amendment No.'s 224 and 225, Limiting Conditions for Operation, 3.14 Special Operations, A. Inservice Hydrostatic and Leak Testing Operation, (SI File No. PNPS-27Q-206).

12. BWRVIP-05 SER (Final), USNRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report, (TAC No. M93925), July 28, 1998, (SI File No. BWRVIP-01-205P).

13. Letter from J. Boska (NRC) to M. Bellamy (ENGC), Pilgrim Nuclear Power Station – Pilgrim Relief Request No. 28, Relief from ASME Code, Section XI, Examinations of Reactor Pressure Vessel Circumferential Shell Welds (TAC No. MB6074), April 11, 2003, (SI File No. PNPS-27Q-208).

14. SI Calculation PNPS-22Q-302, "N2 Nozzle Evaluation," 2/21/06.

15. BWRVIP-26, "BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26)," **EPRI Proprietary**, EPRI Report TR-107285, December 1996, (SI File No. BWRVIP-01-226P).

16. Engineering Report PNPS-EP-06-00001, Revision 0, "Reactor Vessel Internals Inspection Program," (SI File No. PNPS-27Q-207)

17. Pilgrim Nuclear Power Station Final Safety Analysis Report, Section 3.3, Reactor Vessel Internals Mechanical Design, and Section 4.2, Reactor Vessel and Appurtenances Mechanical Design, (SI File No. PNPS-27Q-209).

Table 1: Beltline Curve A for 54 EFPY with Bias Correction Factor on Fluence [8]

Pressure-Temperature Curve Calculation

(Pressure Test = Curve A)

(NOTE: THE ART_{NDT} includes a calculated bias on fluence of 1.78.)

Inputs:

Plant =	Pilgrim			
Component =	Beltline			
Vessel thickness, t =	5.5312	inches, so √t =	2.352	√inch
Vessel Radius, R =	113.91	inches		
ART_{NDT} =	122.7	°F ======>	54 EFPY	
Cooldown Rate, CR =	0	°F/hr		
K_{IT} =	0.00	ksi*inch$^{1/2}$ (From Appendix G, for cooldown rate above)		
$\Delta T_{1/4t}$ =	0.0	°F (no thermal for pressure test)		
Safety Factor =	1.50	(for pressure test)		
M_m =	2.178	(From Appendix G, for inside surface axial flaw)		
Temperature Adjustment =	0.0	°F		
Height of Water for a Full Vessel =	507.5	inches		
Pressure Adjustment =	18.3	psig (hydrostatic pressure for a full vessel at 70°F)		
Hydro Test Pressure =	1,565	psig		
Flange RT_{NDT} =	10.0	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
70.0	70.0	40.43	26.95	0	70.0	0
70.0	70.0	40.43	26.95	601	70.0	583
75.0	75.0	41.19	27.46	612	75.0	594
80.0	80.0	42.03	28.02	625	80.0	606
85.0	85.0	42.95	28.64	638	85.0	620
90.0	90.0	43.98	29.32	654	90.0	635
95.0	95.0	45.11	30.08	671	95.0	652
100.0	100.0	46.37	30.91	689	100.0	671
105.0	105.0	47.75	31.84	710	105.0	691
110.0	110.0	49.28	32.86	733	110.0	714
115.0	115.0	50.97	33.98	758	115.0	739
120.0	120.0	52.84	35.23	785	120.0	767
125.0	125.0	54.91	36.61	816	125.0	798
130.0	130.0	57.19	38.13	850	130.0	832
135.0	135.0	59.72	39.81	888	135.0	869
140.0	140.0	62.51	41.67	929	140.0	911
145.0	145.0	65.59	43.73	975	145.0	957
150.0	150.0	68.99	46.00	1026	150.0	1,007
155.0	155.0	72.76	48.51	1082	155.0	1,063
160.0	160.0	76.92	51.28	1143	160.0	1,125
165.0	165.0	81.52	54.34	1212	165.0	1,193
170.0	170.0	86.60	57.73	1287	170.0	1,269
175.0	175.0	92.21	61.48	1371	175.0	1,352
180.0	180.0	98.42	65.61	1463	180.0	1,445
185.0	185.0	105.28	70.19	1565	185.0	1,547
190.0	190.0	112.86	75.24	1678	190.0	1,659
195.0	195.0	121.24	80.83	1802	195.0	1,784

Table 2: PNPS ART Calculations for 54 EFPY with Bias Correction Factor on Fluence [9]

Pilgrim RPV Material ART Calculations *(54 EFPY)*

(NOTE: This table covers all RPV materials with an exposed fluence, E > 1 MeV, of greater than $1.0x10^{17}$ n/cm^2.) *includes 1.78 calculated bias on fluence*

	Description	Piece No.	Code No.	Heat No.	Estimated Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	Adjustments For 1/4t: ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_I (°F)	ART_{NDT} (°F)
PLATES	Lower Shell #1	337-01A	G-3109-2	C-2957-2	0	0.10	0.47	65.0	30.5	15.3	0.0	61.1
	Lower Shell #2	337-01B	G-3109-1	C-2957-1	-3	0.10	0.48	65.0	30.5	15.3	0.0	58.1
	Lower Shell #3	**337-01C**	**G-3109-3**	**C-2973-1**	**-4**	**0.11**	**0.63**	**74.5**	**35.0**	**17.0**	**0.0**	**65.0**
	Lower-Int. Shell #1	337-03A	G-3108-3	C-2945-2	-12	0.10	0.66	65.6	34.3	17.0	0.0	56.3
	Lower-Int. Shell #2	337-03B	G-3108-1	C-2921-2	-30	0.14	0.60	100.0	52.2	17.0	0.0	56.2
	Lower-Int. Shell #3	337-03C	G-3108-2	C-2945-1	-7	0.10	0.65	65.5	34.2	17.0	0.0	61.2

	Description	Seam No.	Heat No.	Flux Type & Lot No.	Estimated Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	Adjustments For 1/4t: ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_I (°F)	ART_{NDT} (°F)
WELDS	**L. Int. Shell Long. Weld #1**	**1-338A**	**27204/12008**	**Linde 1092 #3774**	**-48**	**0.219**	**0.996**	**231.1**	**114.7**	**28.0**	**0.0**	**122.7**
	L. Int. Shell Long. Weld #2	1-338B	27204/12008	Linde 1092 #3774	-48	0.219	0.996	231.1	78.4	28.0	0.0	86.4
	L. Int. Shell Long. Weld #3	**1-338C**	**27204/12008**	**Linde 1092 #3774**	**-48**	**0.219**	**0.996**	**231.1**	**114.7**	**28.0**	**0.0**	**122.7**
	L. Int./L. Shell Girth Weld	1-344	21935	Linde 1092 #3869	-50	0.183	0.704	172.2	75.4	28.0	0.0	81.4
	Lower Shell Long. Weld #1	2-338A	27204	Linde 1092 #3714	-34	0.203	1.018	226.8	83.5	28.0	0.0	105.5
	Lower Shell Long. Weld #2	2-338B	27204	Linde 1092 #3714	-34	0.203	1.018	226.8	96.5	28.0	0.0	118.5
	Lower Shell Long. Weld #3	2-338C	27204	Linde 1092 #3714	-34	0.203	1.018	226.8	87.8	28.0	0.0	109.8

Fluence Information (see Note 2):

Calculated Fluence Bias = 1.78

	Location	Wall Thickness (Inches) Full [(3)]	Wall Thickness (Inches) 1/4t	Fluence at ID (n/cm^2)	Attenuation, 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
PLATES	Lower Shell #1	5.531	1.383	1.80E+18	0.718	1.29E+18	0.470
	Lower Shell #2	5.531	1.383	1.80E+18	0.718	1.29E+18	0.470
	Lower Shell #3	5.531	1.383	1.80E+18	0.718	1.29E+18	0.470
	Lower-Int. Shell #1	5.531	1.383	2.28E+18	0.718	1.63E+18	0.522
	Lower-Int. Shell #2	5.531	1.383	2.28E+18	0.718	1.63E+18	0.522
	Lower-Int. Shell #3	5.531	1.383	2.28E+18	0.718	1.63E+18	0.522
WELDS	L. Int. Shell Long. Weld #1	5.531	1.383	2.03E+18	0.718	1.46E+18	0.496
	L. Int. Shell Long. Weld #2	5.531	1.383	9.20E+17	0.718	6.60E+17	0.339
	L. Int. Shell Long. Weld #3	5.531	1.383	2.03E+18	0.718	1.46E+18	0.496
	L. Int./L. Shell Girth Weld	5.531	1.383	1.55E+18	0.718	1.11E+18	0.438
	Lower Shell Long. Weld #1	5.531	1.383	1.08E+18	0.718	7.77E+17	0.368
	Lower Shell Long. Weld #2	5.531	1.383	1.45E+18	0.718	1.04E+18	0.425
	Lower Shell Long. Weld #3	5.531	1.383	1.20E+18	0.718	8.58E+17	0.387

Notes:
1. Material information taken from SIA Report No. SIR-00-082, Revision 0, "Updated Evaluation of Reactor Pressure Vessel Materials Properties for Pilgrim Nuclear Power Station," August 2000, Tables 3-1 through 3-12.
2. Fluence values from Transware Report No. ENT-FLU-001-R-001, Revision 0, "Pilgrim Nuclear Power Station Reactor Pressure Vessel Fluence Evaluation," Tables 7-3 and 7-4, and are multiplied by a calculated bias of 1.78.
3. RPV minimum thickness = 5 17/32" per Section 3.3.2 of SIR-00-082, Revision 0.

Table 3. PNPS Charpy Upper Shelf Energy Values for 54 EFPY (Without 1.78 Bias Correction Factor on Fluence) [6]

Material Description						54 EFPY Projection		
Reactor Vessel Beltline Region Location	Matl Type	Material Identification	Heat #	%Cu	Unirradiated CvUSE	1/4 T fluence (10^{19} n/cm^2)	% Drop in USE	USE (1/4 T)
Lower Intermediate Shell	A533B	G-3108-1	C-2921-2	0.14	81	0.084	12.79%	70.6
Lower Intermediate Shell	A533B	G-3108-2	C-2945-1	0.10	80	0.084	10.57%	71.5
Lower Intermediate Shell	A533B	G-3108-3	C-2945-2	0.10	81	0.084	10.57%	72.4
Lower Shell	A533B	G-3109-1	C-2957-1	0.10	76	0.061	9.79%	68.6
Lower Shell	A533B	G-3109-2	C-2957-2	0.10	79	0.061	9.79%	71.3
Lower Shell	A533B	G-3109-3	C-2973-1	0.11	72	0.061	10.31%	64.6
Lower Int/Lower Shell Circ Weld	Linde 1092	1-334	21935	0.18	75	0.057	16.39%	62.7
Lower Int Shell Axial Welds	Linde 1092	1-338A,B,C	27204-12008	0.22	75	0.076	19.52%	60.4
Lower Shell Axial Welds	Linde 1092	2-338A,B,C	27204	0.20	75	0.050	16.87%	62.3

Table 4. PNPS Charpy Upper Shelf Energy Values for 54 EFPY (With 1.78 Bias Correction Factor on Fluence)

Material Description						54 EFPY Projection (with 1.78 bias CF on fluence)		
Reactor Vessel Beltline Region Location	Matl Type	Matl Ident.	Heat#	%Cu	Unirr. CvUSE	1/4t fluence (10^19 n/cm2)	% Drop in USE	USE @ 1/4t
Lower Intermediate Shell	A533B	G-3108-1	C-2921-2	0.14	81	0.129	14.3	69.4
Lower Intermediate Shell	A533B	G-3108-2	C-2945-1	0.10	80	0.129	11.7	70.6
Lower Intermediate Shell	A533B	G-3108-3	C-2945-2	0.10	81	0.129	11.7	71.5
Lower Shell	A533B	G-3109-1	C-2957-1	0.10	76	0.163	12.3	66.7
Lower Shell	A533B	G-3109-2	C-2957-2	0.10	79	0.163	12.3	69.3
Lower Shell	A533B	G-3109-3	C-2973-1	0.11	72	0.163	13.1	62.6
Lower Int./Lower Shell Circ. Weld	Linde 1092	1-334	21935	0.183	75	0.111	19.6	60.3
Lower In. Shell Axial Welds	Linde 1092	1-338A,B,C	27204/12008	0.219	75	0.146	23.2	57.6
Lower Shell Axial Welds	Linde 1092	2-338A,B,C	27204	0.203	75	0.104	20.5	59.6

Table 5. PNPS Maximum Projected Fluence and USE Drop for Vessel Beltline Materials

Material Description					Maximum Projected Fluence and USE Drop			
Reactor Vessel Beltline Region Location	Matl Type	Matl Ident.	%Cu	Unirr. CvUSE	1/4t fluence (10^19 n/cm2)	Max. % Drop in USE	Min. USE @ 1/4t	Fluence Ratio
Lower Intermediate Shell	A533B	G-3108-1	0.14	81	> 6.0	38.3	50.0	> 46.5
Lower Intermediate Shell	A533B	G-3108-2	0.10	80	> 6.0	37.5	50.0	> 46.5
Lower Intermediate Shell	A533B	G-3108-3	0.10	81	> 6.0	38.3	50.0	> 46.5
Lower Shell	A533B	G-3109-1	0.10	76	> 6.0	34.2	50.0	> 36.8
Lower Shell	A533B	G-3109-2	0.10	79	> 6.0	36.7	50.0	> 36.8
Lower Shell	A533B	G-3109-3	0.11	72	> 6.0	30.6	50.0	> 36.8
Lower Int./Lower Shell Circ. Weld	Linde 1092	1-334	0.183	75	1.11	33.3	50.0	10
Lower Int. Shell Axial Welds	Linde 1092	1-338A,B,C	0.219	75	0.71	33.3	50.0	**4.86***
Lower Shell Axial Welds	Linde 1092	2-338A,B,C	0.203	75	0.86	33.3	50.0	8.3

* limiting fluence ratio to reach 50 ft-lbs C_VUSE = (0.71E19)/(0.146E19) = 4.86

Table 6. Effects of Irradiation on RPV Axial Weld Properties

Limiting Axial Welds - Lower Int. Long. Welds #1 and #3

Wire Heat/Lot (27204/12008, Lot No. 3774)

Plant	CE (CEOG)	PNPS	PNPS with Bias CF	PNPS Limit
Parameter Description	USNRC Limiting Plant-Specific Data	Data for axial weld (no bias CF)	Data for axial weld (1.78 bias CF)	Data for axial weld (limiting fluence)
EFPY	64	54	54	>54
Initial (unirradiated) reference temperature (RTndt), °F	0	-48	-48	-48
Neutron fluence at the end of the requested relief period (Peak Surface Fluence in the Beltline), n/cm^2	4.00E+18	1.14E+18	2.03E+18	8.48E+18*
FF = Fluence factor (calculated per Reg. Guide 1.99, Rev. 2)	0.746	0.444	0.573	0.954
Weld Copper content, wt. %	0.219	0.219	0.219	0.219
Weld Nickel content, wt%	0.996	0.996	0.996	0.996
CF = Chemistry Factor	231.1	231.1	231.1	231.1
Increase in reference temperature (ΔRTndt), °F (= FF*CF)	172.4	102.9	132.4	220.4
Mean adjusted reference temperature (ART), °F (= RTndt +ΔRTndt)	172.4	54.9	84.4	172.4

*Fluence ratio = (8.48E18)/(2.03E18) = 4.18

Table 7. Effects of Irradiation on RPV Circumferential Weld Properties

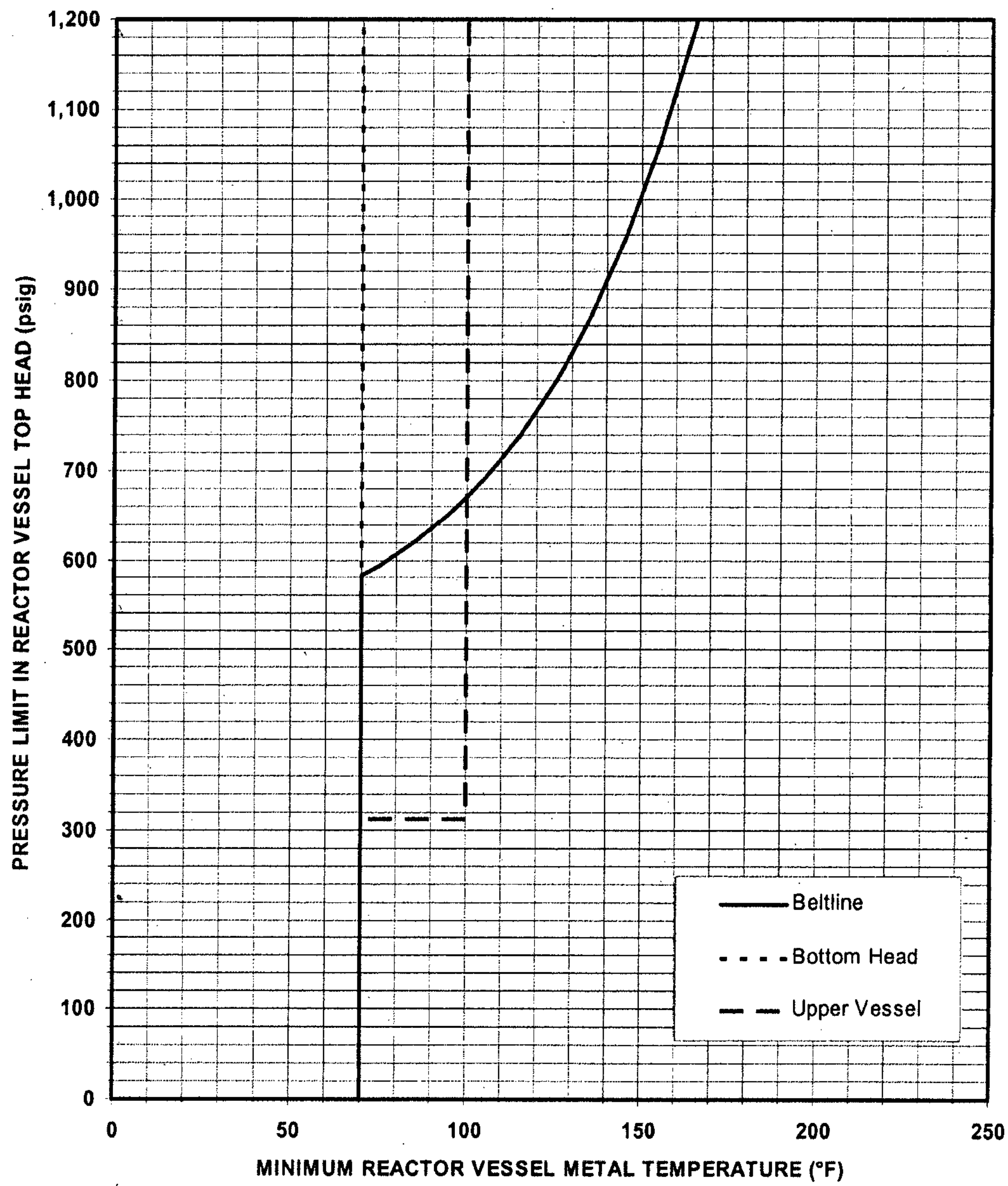
Limiting Circ. Weld – Lower Int.-to-Lower Shell Circ. Weld 1-344

Wire Heat/Lot (21935, Lot No. 3869)

Plant	CE (CEOG)	PNPS	PNPS with Bias CF	PNPS Limit
Parameter Description	USNRC Limiting Plant-Specific Data	Data for circ. weld (no bias CF)	Data for circ. weld (1.78 bias CF)	Data for circ. weld (limiting fluence)
EFPY	64	54	54	>54
Initial (unirradiated) reference temperature (RTndt), °F	0	-50	-50	-50
Neutron fluence at the end of the requested relief period (Peak Surface Fluence in the Beltline), n/cm^2	4.00E+18	8.69E+17	1.55E+18	1.14E+19*
FF = Fluence factor (calculated per Reg. Guide 1.99, Rev. 2)	0.746	0.389	0.510	1.037
Weld Copper content, wt. %	0.183	0.183	0.183	0.183
Weld Nickel content, wt%	0.704	0.704	0.704	0.704
CF = Chemistry Factor	172.2	172.2	172.2	172.2
Increase in reference temperature (ΔRTndt), °F (= FF*CF)	128.5	67.1	87.9	178.5
Mean adjusted reference temperature (ART), °F (= RTndt +ΔRTndt)	128.5	17.1	37.9	128.5

*Fluence ratio = (1.14E19)/(1.55E18) = 7.35

Figure 1: Pressure Test P-T Curve (Curve A) for 54 EFPY with Bias CF on Fluence [8]



PNPS Pressure Test Curve (Curve A), 54 EFPY

(NOTE: The fluence used on the beltline curve is increased by a calculated bias on fluence of 1.78.)

Figure 2: Calculated Hydrotest Temperature and 1/4t ART versus Fluence

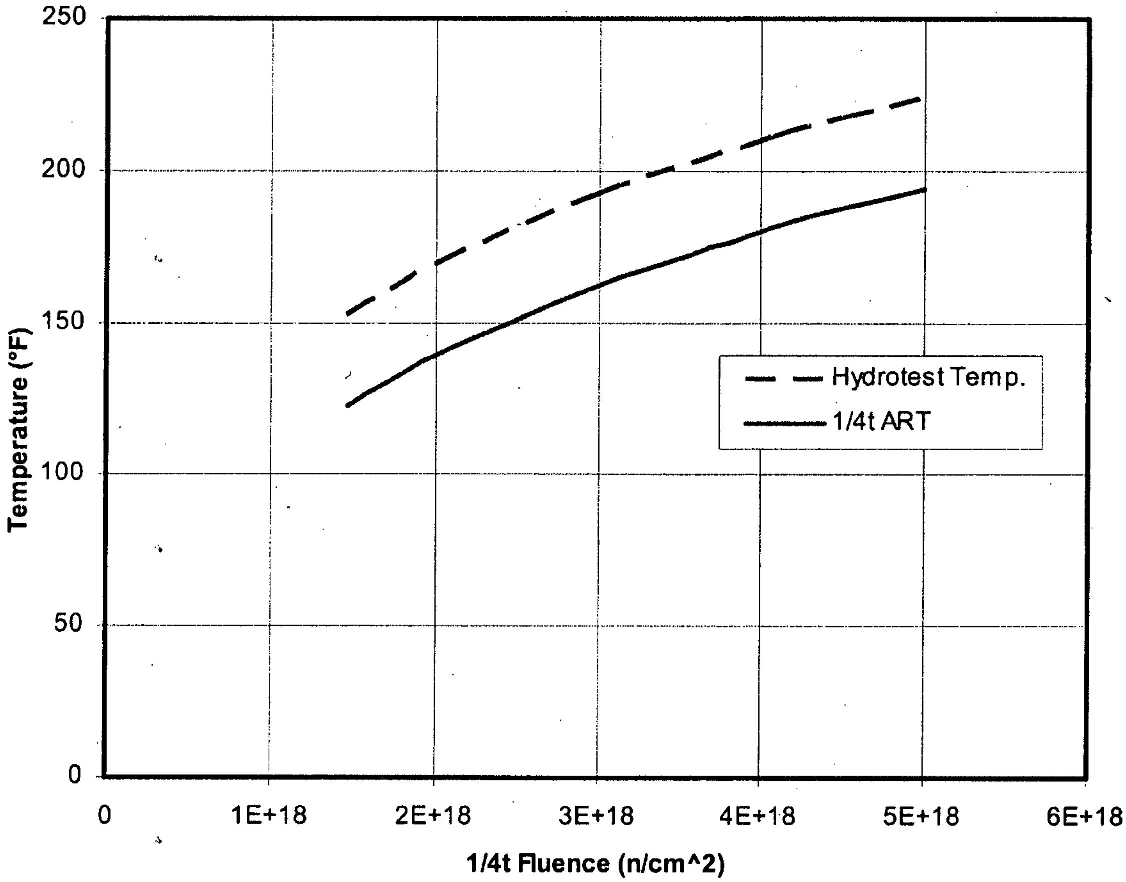


Figure 3. Predicted Decrease in Upper Shelf Energy as a Function of Pct. Copper and Fluence (from Reg. Guide 1.99, Rev. 2 [7])