



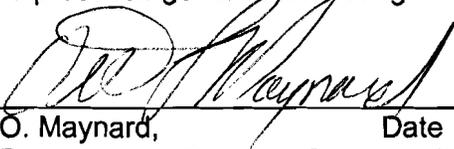
UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

MEMORANDUM TO: Maitri Banerjee, Senior Staff Engineer, ACRS

FROM: O. Maynard, Chairman, Plant License Renewal Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE
SUBCOMMITTEE ON PLANT LICENSE RENEWAL REGARDING
PILGRIM NUCLEAR POWER STATION ON APRIL 4, 2007, IN
ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on April 4, 2007, are an accurate record of the proceedings for that meeting.


O. Maynard, Date 5/4/07
Plant License Renewal Subcommittee Chairman



UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

May 4, 2007

MEMORANDUM TO: O. Maynard, Chairman, Plant License Renewal Subcommittee

FROM: Maitri Banerjee, Senior Staff Engineer, ACRS *Maitri B.S.*

SUBJECT: THE MINUTES OF THE MEETING OF THE SUBCOMMITTEE ON
PLANT LICENSE RENEWAL REGARDING PILGRIM NUCLEAR
POWER STATION ON APRIL 4, 2007, IN ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them at your earliest convenience. If you are satisfied with these minutes please sign, date, and return the attached certification letter.

Attachments: Certification Letter
Minutes

cc w Attachments: Pilgrim License Renewal Subcommittee Members

cc w/o Attachments: F. Gillespie
C. Santos
S. Duraiswami

For the Nuclear Regulatory Commission.
Siva P. Lingam,
Project Manager, Plant Licensing Branch II-1, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.
 [FR Doc. E7-5340 Filed 3-22-07; 8:45 am]
BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

Request for a License To Import Radioactive Waste

Pursuant to 10 CFR 110.70(C) "Public notice of receipt of an application,"

please take notice that the Nuclear Regulatory Commission (NRC) has received the following request for an import license. Copies of the request are available electronically through ADAMS and can be accessed through the Public Electronic Reading Room (PERR) link <http://www.nrc.gov/reading-rm.html> at the NRC Homepage.

A request for a hearing or petition for leave to intervene may be filed within 30 days after publication of this notice in the **Federal Register**. Any request for hearing or petition for leave to intervene shall be served by the requestor or petitioner upon the applicant, the Office

of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555; the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555; and the Executive Secretary, U.S. Department of State, Washington, DC 20520.

The information concerning this import license application follows.

NRC IMPORT LICENSE APPLICATION

Name of applicant, date of application, date received, application No. docket No.	Description of material		End use	Country of origin
	Material type	Total quantity		
Westinghouse Electric Company LLC, February 28, 2007, March 1, 2007, IW021, 11005674.	Contaminated materials (waste filter cake and shot) generated by Mississauga Metals & Alloys in Ontario, Canada from the decontamination of steel received from the Hematite facility in Festus, MO under NRC License XW003.	Mississauga is required to return up to 74,843.0 kilograms of filter cake and 122,470.0 kilograms of shot, which it expects to generate from the decontamination process. It is expected that these materials will be contaminated with up to 3.506 kilograms uranium-235 contained in up to a total of 72.288 kilograms of uranium with a maximum enrichment of 4.9 weight percent.	All materials are to be disposed of at the Energy Solutions of Utah site located in Clive, Utah in accordance with an existing agreement between the parties.	Canada.

Dated this 13th day of March 2007 at Rockville, Maryland.
 For the Nuclear Regulatory Commission.
Margaret M. Doane,
Deputy Director, Office of International Programs.
 [FR Doc. E7-5336 Filed 3-22-07; 8:45 am]
BILLING CODE 7590-01-P

will hear presentations by and hold discussions with representatives of the NRC staff, Entergy Nuclear Operations, Inc., and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Ms. Maitri Banerjee (telephone 301/415-6973) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 6:45 a.m. and 3:30 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: March 15, 2007.
Cayetano Santos,
Acting Branch Chief, ACRS.
 [FR Doc. E7-5335 Filed 3-22-07; 8:45 am]
BILLING CODE 7590-01-P

RAILROAD RETIREMENT BOARD

Proposed Collection; Comment Request

Summary: In accordance with the requirement of Section 3506(c)(2)(A) of the Paperwork Reduction Act of 1995, which provides opportunity for public comment on new or revised data collections, the Railroad Retirement Board (RRB) will publish periodic summaries of proposed data collections.

Comments are invited on: (a) Whether the proposed information collection is necessary for the proper performance of the functions of the agency, including whether the information has practical utility; (b) the accuracy of the RRB's estimate of the burden of the collection of the information; (c) ways to enhance the quality, utility, and clarity of the information to be collected; and (d) ways to minimize the burden related to

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS); Meeting of the Subcommittee on Plant License Renewal; Notice of Meeting

The ACRS Subcommittee on Plant License Renewal will hold a meeting on April 4, 2007, Room T-2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance. The agenda for the subject meeting shall be as follows:

Wednesday, April 4, 2007—10:30 a.m. until 5 p.m.

The purpose of this meeting is to review the Pilgrim license renewal application and the associated Safety Evaluation Report. The Subcommittee

**Advisory Committee on Reactor Safeguards
Plant License Renewal Subcommittee Meeting
Pilgrim Nuclear Power Station
April 4, 2007
Rockville, MD**

-PROPOSED SCHEDULE-

Cognizant Staff Engineer: Maitri Banerjee mxb@NRC.GOV (301) 415-6973

Topics	Presenters	Time
Opening Remarks	O. Maynard, ACRS	10:30am - 10:35 am
Staff Introduction	P. T. Kuo, NRR	10:35 am - 10:40 am
Introduction and Background	Entergy	10:40 pm - 11:00 am
Open Items	Entergy	11:00 am - 12:00 pm
Lunch		12:00 pm - 1:00 pm
Completion of Open Item Discussion	Entergy	1:00 pm - 1:30 pm
NRC Staff Presentation		1:30 pm - 2:30 pm
A. Scoping and Screening Results	Perry Buckberg, NRR	
B. Onsite Inspection Results	Glenn Meyer, Region I Richard Conte, Region I	
C. NRC Audit	James Davis, NRR	
D. Time Limited Aging Analyses	Perry Buckberg, NRR	
E. Open Items	Perry Buckberg, NRR	
Break		2:30 pm - 2:45 pm
Public Comment (if any requested)		2:45 pm - 3:00 pm
Subcommittee Discussion	O. Maynard, ACRS	3:00 pm-4:00 pm

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.
- Entergy is the licensee for Pilgrim Nuclear Power Station and the applicant for a license renewal.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

APRIL 4, 2007
Date

PLEASE PRINT

	<u>NAME</u>	<u>ORGANIZATION</u>
1	David Lach	Entergy
2	Bryan Ford	Entergy
3	Ed Sanchez	Entergy
4	MIKE FOLLIN	Constellation Energy
5	Steve Bethay	Entergy
6	FRED MOGOLESKO	ENERGY
7	Brian R. Sullivan	Entergy
8	Franz - Joseph WLM	MIT
9	John Hoffman	Entergy (NY)
10	BARRY M. GORDON	Structural Integrity Associates
11	Alec Alizer	Entergy (JRF)
12	Jim Costedio	Entergy (JRF)
13	John McCann	ENERGY
14	JOHN DREIFUSS	ENERGY
15	Tim Griesbach	Structural Integrity Associates
16	Tom White	Entergy
17	Raymond Pace	Entergy
18	MIKE STRAUD	ENERGY
19	DAVID MANNAS	ENERGY
20	A Michael Metell	Entergy

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

APRIL 4, 2007
Date

PLEASE PRINT

	<u>NAME</u>	<u>ORGANIZATION</u>
1	<u>ROBERT SMITH</u>	<u>ENERGY</u>
2	<u>Gary Dyckman</u>	<u>"</u>
3	<u>REZA AHRABLI</u>	<u>"</u>
4	<u>TED Juy</u>	<u>ENERGY</u>
5	<u>ALAN COX</u>	<u>ENERGY</u>
6	<u>GARRY G. YOUNG</u>	<u>ENERGY</u>
7	<u>JAY K. THAYER</u>	<u>ENERGY/NEI</u>
8	<u>CISS MARKS</u>	<u>ISL</u>
9	<u>MIKE GALBREATH</u>	<u>EXELON</u>
10	<u>WEN-CHUN TOKO</u>	<u>AEC, TAIWAN</u>
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

APRIL 4, 2007
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	<u>NAME</u>	<u>NRC ORGANIZATION</u>
1	JAMES MEDOFF	NEC/NRR/DLR
2	Glenn Meyer	Region I
3	DAN Hunter	NRR/DLR
4	Jim DAVIS	NRR/DLR
5	Perry Buckberg	NRR/DLR
6	Robert Schaub	NRR/DLR
7	DUC NGUYEN	NRR/DLR
8	Ed Smith	NRR/ SS DSS
9	Lambros Lois	NRR/DSS/SRFB
10	ANGELO STUBBS	NRR/DSS/SBPB
11	RAJ AULUCK	NRR/DLR
12	K. Robert HSU	NRR/DLR
13	Ken Chang	NRR/DLR
14	Yeon-Ki Chung	NRR/DLR
15	Roy MATHEW	NRR/DLR.
16	SURINDER ARORA	NRR/DLR
17	Bryce Lehman	NRR/DLR
18	EVAN DAVIDSON	NRR/DSS
19	Louise Lund	NRR/DLR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

APRIL 4, 2007
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	<u>NAME</u>	<u>NRC ORGANIZATION</u>
1	<u>Naeem IQBAL</u>	<u>NRR/DRA/AFPB</u>
2	<u>Veronica Rodriguez</u>	<u>NRR/DLR/RLRB</u>
3	<u>2 Ram Subbaratnam</u>	<u>NRO/DNRL/NHIF</u> <i>NAIF</i>
4	<u>Ganesh Choumali</u>	<u>NRR/DCI/CVIB</u>
5	<u>Matthew A. Mitchell</u>	<u>NRR/DCI/CVIB</u>
6	<u>DAVID ROTH</u>	<u>OGC</u>
7	<u>MAURICE HEATH</u>	<u>NRR/DLR</u>
8	<u>Christopher Sydney</u>	<u>NRR/DCI/CVIB</u>
9	<u>Allison Black</u>	<u>NRR/DCI/CVIB</u>
10	<u>David Jeug</u>	<u>NRO/DE/SEB2</u>
11	<u>Tommy Le</u>	<u>NRR/DLR/RLRB</u>
12	<u>Bill Rogers</u>	<u>NRR/DLR/RLRB</u>
13	<u>Donnie Ashley</u>	<u>NRR/DLR/RLRB</u>
14	<u>Keat Howard</u>	<u>NRR/DLR/RLRB</u>
15	<u>Devenber Reddy</u>	<u>NRR/DLR/RLRB</u>
16	<u>Jim Kim</u>	<u>NRR/DORL</u>
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**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE MEETING OF THE SUBCOMMITTEE ON PLANT LICENSE RENEWAL
REGARDING PILGRIM NUCLEAR POWER STATION ON APRIL 4, 2007,
ROCKVILLE, MARYLAND**

On April 4, 2007, the ACRS Subcommittee on Plant License Renewal held a meeting regarding Pilgrim Nuclear Power Station (Pilgrim) in Room T-2B3, 11545 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss the Pilgrim application for license renewal and NRC staff review of it. In addition to the NRC staff, representatives from Entergy (the Pilgrim operator and the licensee) made presentations to the Committee. The meeting was convened at 10:30 a.m. and adjourned around 3:20 p.m. The meeting was open to the public. No written comments or requests to make oral statements were received from the public related to this meeting. Upon request, a telephone bridge was provided for members of the press to listen in.

Attendees

ACRS Members/Staff	Entergy Presenters	NRC Staff	Entergy/Other
Otto Maynard (Chairman)	G. Dyckman	L. Lois	J. Thayer
Graham Wallis (Member)	A. Cox	A. Stubbs	J. McCann
William Shack (Member)	F. Mogolesko	R. Auluck	T. White
Thomas Kress (Member)	F. Ulm (MIT/Entergy)	K. Hsu	J. Costedio
J. Sam Armijo (Member)	B. Gordon (SIA/Entergy)	D. Ashley	G. Young
Said Abdel-Khalik (Member)	NRC Staff	Y. Chung	J. Dreyfuss
Mario Bonaca (Member)	L. Lund	R. Matthew	
J. Barton (Consultant)	M. Mitchell	S. Arora	M. Stroud
Maitri Banerjee (DFO)	R. Schaaf	B. Lehman	E. Sanchez
NRC Staff Presenters	K. Chang	E. Davidson	D. Lach
P. Buckberg, NRR	J. Medoff	N. Iqbal	D. Mannai
G. Meyer, RI	D. Harrison	C. Sydnor	H. Metell
J. Davis, NRR	D. Roth	J. Kim	R. Ahrabli
Entergy Presenters	M. Heath	E. Smith	T. Ivy
R. Smith	G. Cheruvenki	B. Rogers	M. Fallin
S. Bethay	D. Nguyen	A. Black	J. Hoffman
B. Sullivan	D. Reddy	D. Jeug	C. Marks
B. Ford	V. Rodriguez	T. Le	M. Gallagher
R. Pace	R. Subbaratnam		W. Teng
T. Griesbach	K. Howard		R. Plasse

The presentation slides and handouts used during the meeting are attached to the Office Copy of these minutes. The presentations to the Subcommittee are summarized below.

Chairman Maynard convened the meeting by introducing the ACRS members present. Mr. Maynard stated that the purpose of the meeting was to review the Pilgrim license renewal application, with particular focus on the unresolved items in the staff's draft safety evaluation. He then called upon the NRR Project Manager, Perry Buckberg, to begin the staff presentation. Mr. Buckberg introduced the NRC staff presenters and the management present, and turned over the presentation to the Pilgrim licensee.

Mr. Robert Smith, General Manager for Pilgrim Plant Operations, introduced the Pilgrim team present to the meeting to support their presentation to ACRS. Mr. Bethay, Director of Nuclear Safety Assurance at Pilgrim, started his presentation by describing the general information related to plant location, design, licensing, and upgrades made to the plant over the years. Mr. Bethay stated that the 16th refueling outage starting in a few days has been designated as a license renewal outage to install several plant modifications, upgrades and major equipment replacements. He pointed out that the Pilgrim license renewal application was prepared to address the Revision 1 of the Standard Review Plan for License Renewal and the NRC's Generic Aging Lessons Learned (GALL) report, in addition to in-house and industry lessons learned findings. He stated that the application was peer reviewed by ten utilities in addition to going through the existing Pilgrim safety and quality review processes. Out of the 40 aging management programs Entergy committed to for the period of license extension, 14 existing programs will require enhancement and 10 new programs are required to be developed. Additionally, Entergy is evaluating several cost-beneficial severe accident mitigation alternatives related upgrades for implementation.

Mr Bethay addressed the four open items in the staff's safety evaluation (SE). The first open item involved the security diesel generator and confirming proper scoping of its components in the license renewal program. Mr. Bethay stated that the Region I license renewal inspection reviewed the details and found the scoping, which included all of the security diesel components, to be acceptable.

The second open item involved the inaccessible fire barrier penetration seals, whether all such seals are included in the aging management program. As all such seals are accessible for periodic inspection at Pilgrim, the licensee believes that the open item is resolved pending NRC confirmation.

The third open item in the SE involves the containment and has several issues. The first issue was the potential for corrosion in the inaccessible area of the steel containment shell. Mr. Bethay described the diverse design features provided at Pilgrim to minimize the potential for undetected water intrusion in to the air gap between the containment steel shell and the concrete. He also presented multiple water collection and detection provisions. Entergy has taken steps to ensure these design features are maintained operational, and also verified through inspections (visual, boroscope and limited ultrasonic testing (UT)) that the containment shell is not degrading. Entergy stated that they would continue doing confirmatory UT, in addition to keeping the design provisions operable. The ACRS members asked many probing questions regarding the details of the design, inspections, and licensee's findings.

The other containment open item involved finding of water in the torus room floor. The licensee stated that they had confirmed the source of the water to be ground water coming up around

some of the torus saddle anchor bolt base plates due to hydraulic pressure, and concluded that it has no effect on the integrity of the concrete, embedded steel or anchor bolts in the torus room floor. The hydraulic pressure is caused by the higher water table (about 25 feet above the bottom of the reactor building) due to the geology of the site. The licensee presented various site construction diagrams showing the relative heights of the containment base mat and the ground water level. The licensee has committed to monitor the water, the concrete and the bolts and perform repair of the existing bolt corrosion.

Upon Chairman Maynard's questions, Mr. Bethay indicated that their assessment was based on monitoring of water in the torus room, that comes in and evaporates, and multiple engineering evaluations done over the years. Mr. Bethay showed pictures of various bays in the torus room, with some showing puddles of water, the worst being not of a depth that can be pumped out. Upon questions from the members, Mr. Bethay stated that only a few of the bays have water coming in, and that a small leaching of calcium from the concrete is seen in some of the bays. Upon Dr. Armijo's question, Mr. Bethay also stated that the collection containers from the drain lines in the sand bed region are periodically inspected, and always found dry, hence the containment refueling water drains can not be the source of the torus room water. A chemical analysis of the water also proves it not to be from the refueling pool.

Professor Ulm from MIT was contracted by the licensee to perform an analysis of the base mat concrete condition and cause of the water intrusion. Prof. Ulm stated that uneven heat distribution during construction of the reactor building base mat, when the outer donut structure (torus room base mat) was poured in various sections before the thicker inner core, resulted in stress concentration at the construction joints and ensuing cracks in concrete. He stated the membrane placed under the base mat had failed also. Hence, according to the licensee's analysis, ground water is finding its way through vertical construction joints and localized cracks in concrete and reaching the surface ultimately near the torus saddle support mounting bolts where the concrete is cored out by 2 to 2 1/2 feet. Prof. Ulm calculated that a hypothetical cylinder with a diameter of 1/6th of an inch through the base mat in a torus room bay would result in the amount of water that has been seeping in under the hydraulic pressure.

Dr. Wallis questioned if the ground water seeping through the concrete could corrode the anchor bolts. The licensee explained that some amount of corrosion is seen, but small levels of corrosion increases the bonding between the steel and the concrete. However, as corrosion increases some cracking of the concrete around it could happen. So, they plan to do a more detailed inspection of the bolt corrosion. Regarding the reinforcing bars inside the concrete, given the small amount of leak, it is extremely unlikely, even if the whole leak is assumed to be through one 1/6 inch cylinder, that a reinforcement bar will be degraded by this leak, according to licensee's analysis. Given the lower pH value of about 9.4, measured in the torus room water, compared to the higher pH value of 13 for concrete, the licensee opined that concrete dissolution into the seeping water is occurring very slowly, thus, not to impact the structural integrity of the base mat within the license renewal time frame of concern. The small amount of calcium deposited in the bays where water is seeping in indicates a very localized but a slow process of concrete degradation not relevant to the time scale of concern.

The licensee plans to continue monitoring the water (amount, chemistry etc); and inspect a sample of the bolts and the grout around them to verify that the leakage is coming in from the area around the bolts, check the integrity of the bolts and the grout, and take necessary corrective actions. Upon Dr. Wallis' question regarding adequacy of the sample inspection every 10 years committed by the licensee, Mr. Bethay stated that routine (operator) rounds

would look at the bolts, and any changes identified will be addressed through the licensee's corrective action program.

The fourth open item involved the lack of benchmarking data to support plant specific neutron fluence calculations for use in time limited aging analyses (TLAAs). There was a significant difference between the fluence values indicated by Pilgrim reactor vessel surveillance capsule pulled during the 4th refueling outage and the fluence values predicted by the RAMA computer code. In addition to impacting the neutron embrittlement TLAAs, this problem impacts the modified TLAAs that use the acceptable fluence values as a basis for determining compliance with regulations. These additional TLAAs are the pressure-temperature limits, Charpy upper-shelf energy, adjusted reference temperature, reactor vessel circumferential weld inspection relief, and reactor vessel axial weld failure probability. This problem also impacts the current licensing basis in the pressure temperature curves, but the licensee stated that the current curves are validated through cycle 18 (year 2011 refueling outage) by using a conservative bias to account for the discrepancy. The licensee committed to provide a resolution plan by September 2007, and submit calculations demonstrating compliance with RG 1.190 by June 2010 to support startup from cycle 18. They are planning to pull another capsule during the upcoming 16th refueling outage. This capsule has been in the vessel much longer than the prior capsule pulled. The licensee is going to a sister plant also for additional capsule data, and reviewing the power history and other data input to their calculation, such that they are able to benchmark the RAMA code for Pilgrim before cycle 19. If a problem with the code is identified, the NRC may need to re-perform its review that approved the code.

Dr. Bonaca asked if the industry expected to develop appropriate tools for inspecting inaccessible in-vessel welds on the core spray piping and the jet pump assembly (in near future). The licensee stated that their probabilistic fracture mechanics analyses show the risk of failure for these welds to be within acceptable limits, thus justifying not doing inspections. These analyses have to be redone with higher fluence level for extended operation.

Dr. Bonaca also asked about the status of the service water cables in light of the problems identified with inaccessible medium voltage cables. The licensee stated that accessible portions of same type of cable are inspected at other locations and that industry data have not identified the service water system cables at voltage level of 480 volts (which are not considered as medium voltage) to be susceptible to the water intrusion issue. Mr. Duc Nguyen, NRC staff, stated that the service water cable voltage of concern will be treated as low voltage cables, and that NRC has issued a generic letter requesting information from the licensees regarding surveillance of inaccessible low voltage cables. Resolution of this issue through the Generic Letter process will be carried over to the license renewal period.

Mr. Barton of ACRS questioned the licensee's conclusion in their license renewal application that aluminum components, exposed to the external salt environment, have no environmental aging. Mr. Barton mentioned the experience of plant trips due to salt buildup in switchyard components. The licensee explained that the insulators in the switchyard are replaced with ones with protective coatings and there has not been a plant trip due to salt deposits on insulators. Also they credited the aggressive monitoring, repair and preventive maintenance program for this success. The licensee stated that salt deposits are much affected by weather conditions and not considered an aging effect, and thus independent of license renewal. The licensee also stated that the station blackout diesel exhaust silencer is the only aluminum component of concern in outdoor environment, and it has been found to have no significant degradation to its structural adequacy.

Mr. Barton also questioned the lack of an aging management program for fuse holders. Mr. Nguyen from the staff noted that the NRC had issued guidance regarding this matter in 2001, and considered fuse holders outside the fuse panel to require aging management review. Since Pilgrim has none outside the fuse panel, they are not subject to aging management review. Mr. Barton also questioned the adequacy of one time UT of the bottom of the condensate storage and diesel fuel tanks that sit on sand beds. The licensee pointed out that these tanks are subject to periodic visual inspection for corrosion inside and outside and maintenance review, in addition to the confirmatory UT. Also, these carbon steel tanks are lined inside, with a relining and UT scheduled for 2008.

Dr. Abdel-Khalik asked about potential corrosion of the containment liner below the floor level due to leak from components above the floor. The licensee explained that the concrete floor is bonded to the steel liner, and UT results at points on the junction of the floor to the liner and up to one inch below have shown no degradation. These confirmatory measurements are to be continued in the future. Dr. Abdel-Khalik was concerned about the potential corrosion of the torus wall. The licensee pointed out that they have an active program of inspecting the inside torus wall and doing necessary restoration of the coating every other outage. Although the same exact locations may not be UT measured at the next outage, the licensee stated that the visible nature of the corrosion provides confidence that the program is effective in identifying degradation. Also, the licensee compares data with another plant (Fitzpatrick) of same vintage to ensure the coating is appropriately maintained.

Mr. Perry Buckberg, the NRR license renewal project manager for Pilgrim, provided an overview of the staff's license renewal review, audit and regional inspections at the plant. The audit was performed by a team of NRC staff and contractors. The audit team confirmed the licensee's scoping and screening methodology and the adequacy of the scope of systems and structures within the license renewal program. The open item related to the security diesel was reviewed by the regional inspection and closed.

The ACRS members asked if the clarity of the regulatory guidance regarding the scope of the license renewal program was adequate. This question arose due to the licensee's addition to the scope as a result of staff review. The staff stated that the identification of equipment function and environment had a role in it, and not the clarity of guidance.

Mr. Glenn Meyer, team leader of the Region I inspection team, provided a summary of their inspection regarding the license renewal program scope and implementation of the aging management program. Upon members' questions he stated that the licensee's characterization of water in the torus room matched the finding of his inspection team, and that it is not unusual to find a small amount of moisture in the torus room. Regarding the location of the catch containers (buckets) for the containment refueling water drains, the licensee stated that the buckets are in place since 1987, but one has to crawl under the torus to find them. During refuel outages, the licensee performs a visual inspection of the buckets to confirm that they are dry, and document the results.

Mr. Meyer stated that their inspection found some incorrect boundaries regarding structural interaction between safety and non-safety system components. Once identified, the licensee resolved the issue to the team's satisfaction. The team identified that a flow switch that monitors the containment refueling water leak had failed, which was later repaired. Region I plans to do another round of inspections before the period of extended operation to ensure the

issues and licensee commitments are addressed. Mr. Meyer ended his presentation with a discussion on the licensee's current performance and that it was in the licensee response column. Mr. Meyer also stated that he believed that the licensee's system engineers at Pilgrim are knowledgeable and provides a generally effective program of identifying problems in the field and addressing them.

Dr. Bonaca asked if the GALL report is forcing the use of request for exceptions from the licensees and ensuing staff evaluations that could be avoided if GALL were not so prescriptive. The staff pointed out that they have been tracking and compiling individual use of acceptable exceptions that would be factored into the next update of GALL, thus making exception requests unnecessary. The staff has been made aware of this list for use in license renewal application review and audits.

Dr. Jim Davis presented the findings of the audit team. Dr. Davis pointed out that although many of the licensee's commitment implementation dates are coincident with the end of the current license, many of the commitments are expected to be completed prior to that date. As there are no inaccessible fire seals at Pilgrim, that open item was readily closed. Dr. Shack questioned if at least 10 percent of each type of fire seals are being inspected. The licensee pointed out that given the large number of each type of seals in the program, a random selection of 20 percent of all seals is expected to include at least 10 percent of each type.

The members asked for earthquake experience at the plant, and the licensee pointed out that the year 1860 earthquake was the plant's design basis. Dr. Ulm stated that he had done an analysis to determine the effect of the earthquake on the reactor building base mat, and the results show slightly higher water intrusion in the torus room, by about 33 percent (or six liters) over the normal operating conditions and no damage to the structural performance of the base mat.

In his presentation regarding closure of the open item on neutron fluence, Mr. Buckberg pointed out that the licensee is expected to propose fluence limits regarding the affected TLAA's and confirm adequacy of those limits via an acceptable method. The staff would impose a license condition to ensure adequate resolution of this open item prior to the period of extended operation.

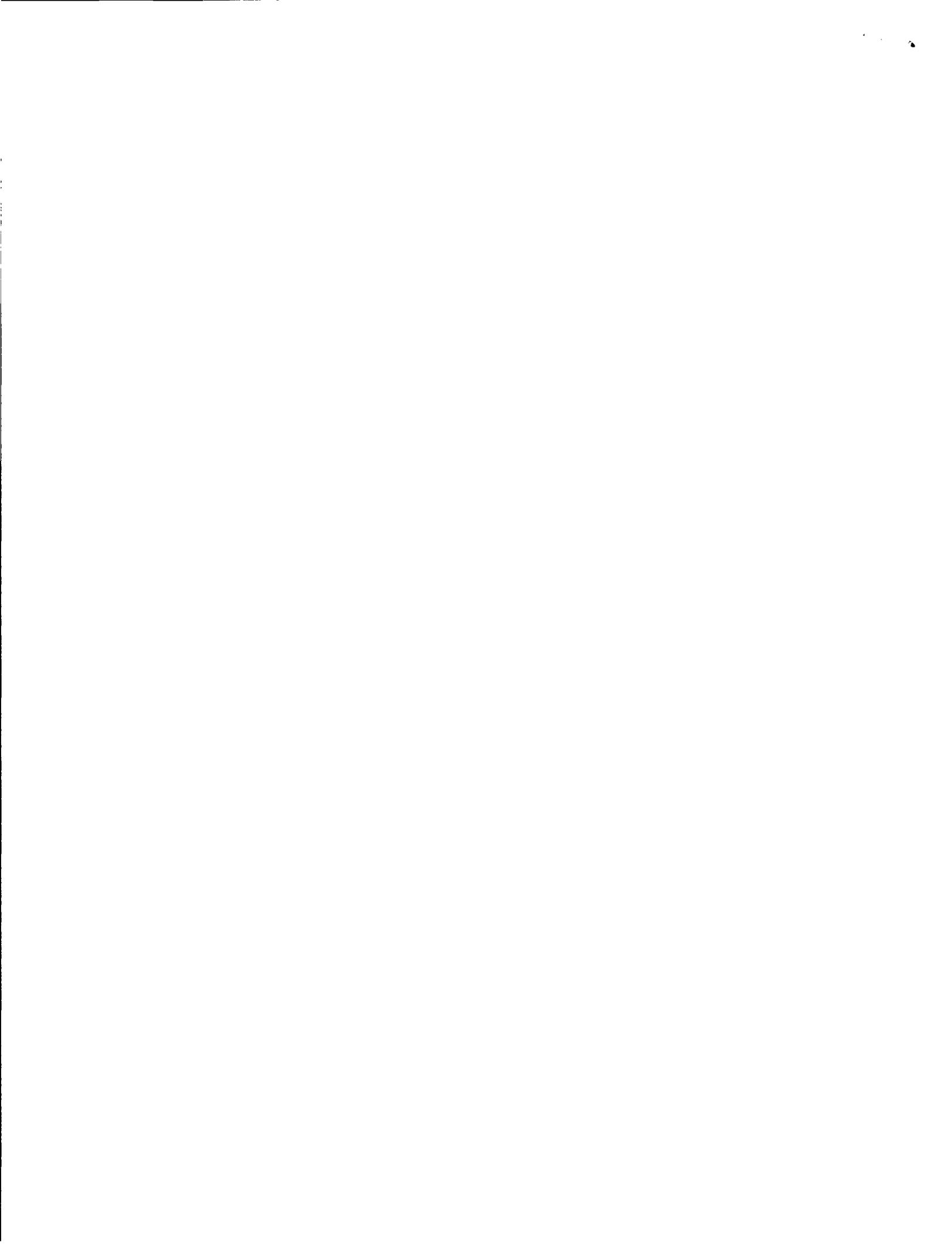
Chairman Maynard then proceeded to identify items that would need to be discussed at the upcoming Full Committee meeting in September. He identified the neutron fluence issue and the torus room water intrusion issue to be definite items for the meeting. Dr. Abdel-Khalik stated that a more realistic calculation for the amount of water leakage may be helpful. Dr. Shack asked if any attempt has been made to identify the exact locations with the amount of water getting in. The licensee pointed to one of the pictures of bay no. 10 that showed a tented enclosure. The licensee stated that a berm was built around one of the anchor bolts and tented to capture the leak without any other condensation dripping into it. The location was completely dried and the licensee recorded the time it took for water to reappear there. Their calculation showed a leakage rate higher than that would be for the theoretical permeability of concrete, thus proving the existence of a flow path.

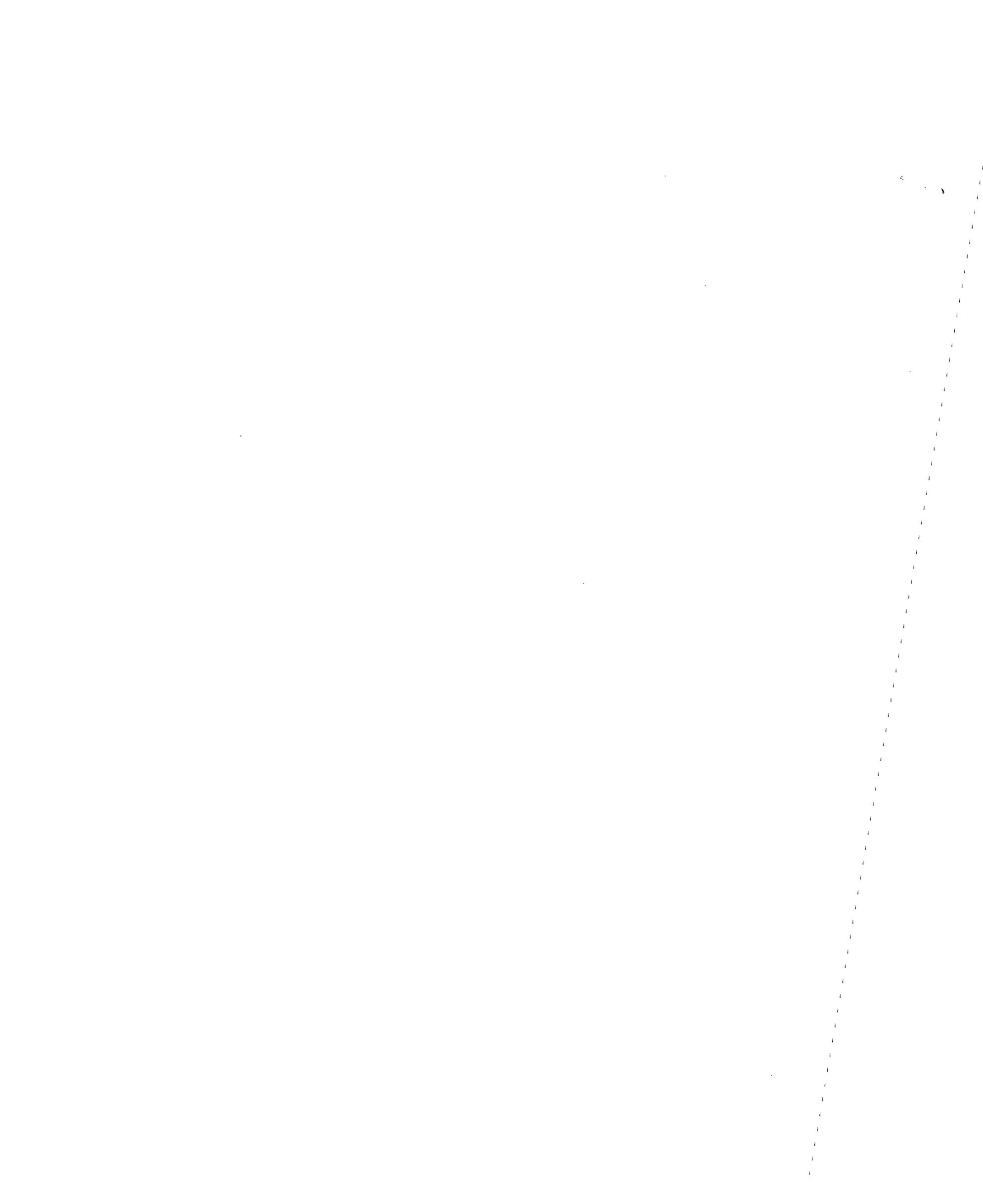
Dr. Bonaca pointed out that in addition to the two issues mentioned above, a presentation on the license renewal inspection may be of interest to the Full Committee. Dr. Abdel Khalik stated that he would like to know about the licensee's processes for performing the benchmarking of the fluence code for Pilgrim (in more detail). Dr. Lois of the staff indicated that the staff had

experienced similar problems before (problem with accurately identifying the location of the surveillance capsule after it is pulled, given the steep gradient of the fluence value with location), and was able to successfully resolve them. Dr. Armijo and Dr. Abdel-Khalik asked that the staff include their experience in resolving similar prior problems in their presentation to the Full Committee. Although Dr. Shack felt that the staff appeared to have a well defined process for resolving the fluence issue, Dr. Kress wondered why other BWR results could not be used to validate the RAMA code for Pilgrim fluence. Dr. Lois from the staff answered that the type of the reactor makes a big difference for BWRs, and Pilgrim is the first BWR3 providing only one data point. Mr. Barton of the ACRS stated that the licensee's application was well prepared in that it was easy to follow.

Dr. Wallis noted that he would have expected the licensee to clarify the torus room water issue in their application for license renewal, as it is a long standing issue, and not wait for an NRC inspection to identify it. Chairman Maynard and Mr. Barton noted that it is not uncommon to find water in some areas of the plant like the torus room (writer's note: they were probably hinting at condensation from the air due to lower area temperature and infrequent but potential small pipe leaks). Dr. Davis of the staff pointed out that the interim staff guidance on containment shell degradation requires an inspection of the torus room for water, thus making the staff more sensitive to the issue of water intrusion. Ms. Lund from NRR management noted the value of a license renewal inspection.

Chairman Maynard adjourned the meeting by thanking everyone attending the meeting.





From: Maitri Banerjee
To: omaynard@allegiance.tv
Date: 05/01/2007 1:14:25 PM
Subject: Fwd: Minutes of the Pilgrim License Renewal Subcommittee Meeting on April 4, 2007

>>> Maitri Banerjee 05/01/2007 1:05 PM >>>
Hello Otto:

I have attached a copy of the draft meeting minutes for your review and comment, and eventual certification. Please let me know if you want to make any changes. I apologize for the delay.

I put the Subcommittee members present at the meeting in cc, and asking them to let me know if they see any mistakes or problems in how an issue may have been characterized, or if they want any changes.

Thank you so much.

With best regards.

Maitri

Mail Envelope Properties (46377571.870 : 12 : 35638)

Subject: Fwd: Minutes of the Pilgrim License Renewal Subcommittee Meeting on
April 4, 2007

Creation Date 05/01/2007 1:14:25 PM

From: Maitri Banerjee

Created By: MXB@nrc.gov

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allegiance.tv PM	Transferred	05/01/2007 1:14:41
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MESSAGE	1197	05/01/2007 1:14:25 PM
Mtg Summary April 4 07.wpd	318679	05/01/2007 12:16:10 PM

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Return Notification: None

Concealed Subject: No
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From: Maitri Banerjee
To: omaynard@alligiance.tv
Date: 05/01/2007 1:05:04 PM
Subject: Fwd: Minutes of the Pilgrim License Renewal Subcommittee Meeting on April 4, 2007

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Thank you so much.

With best regards.

Maitri

CC: Cayetano Santos; graham.b.wallis@dartmouth.edu; happymariner@adelphia.net; jsarmijo@msn.com; mvbonaca@snet.net; said.abdelkhalik@me.gatech.edu; Sam Duraiswamy; tskress@aol.com; wjshack@anl.gov



UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

May X, 2007

MEMORANDUM TO: O. Maynard, Chairman, Plant License Renewal Subcommittee

FROM: Maitri Banerjee, Senior Staff Engineer, ACRS

SUBJECT: THE MINUTES OF THE MEETING OF THE SUBCOMMITTEE ON
PLANT LICENSE RENEWAL REGARDING PILGRIM NUCLEAR
POWER STATION ON APRIL 4, 2007, IN ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them at your earliest convenience. If you are satisfied with these minutes please sign, date, and return the attached certification letter.

Attachments: Certification Letter
Minutes

cc w Attachments: Pilgrim License Renewal Subcommittee Members

cc w/o Attachments: F. Gillespie
C. Santos
S. Duraiswami

DRAFT (Issued 05/01/07)

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE MEETING OF THE SUBCOMMITTEE ON PLANT LICENSE RENEWAL
REGARDING PILGRIM NUCLEAR POWER STATION ON APRIL 4, 2007,
ROCKVILLE, MARYLAND**

On April 4, 2007, the ACRS Subcommittee on Plant License Renewal held a meeting regarding Pilgrim Nuclear Power Station (Pilgrim) in Room T-2B3, 11545 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss the Pilgrim application for license renewal and NRC staff review of it. In addition to the NRC staff, representatives from Entergy (the Pilgrim operator and the licensee) made presentations to the Committee. The meeting was convened at 10:30 a.m. and adjourned around 3:20 p.m. The meeting was open to the public. No written comments or requests to make oral statements were received from the public related to this meeting. Upon request, a telephone bridge was provided for members of the press to listen in.

Attendees

ACRS Members/Staff	Entergy Presenters	NRC Staff	Entergy/Other
Otto Maynard (Chairman)	G. Dyckman	L. Lois	J. Thayer
Graham Wallis (Member)	A. Cox	A. Stubbs	J. McCann
William Shack (Member)	F. Mogolesko	R. Auluck	T. White
Thomas Kress (Member)	F. Ulm (MIT/Entergy)	K. Hsu	J. Costedio
J. Sam Armijo (Member)	B. Gordon (SIA/Entergy)	D. Ashley	G. Young
Said Abdel-Khalik (Member)	NRC Staff	Y. Chung	J. Dreyfuss
Mario Bonaca (Member)	L. Lund	R. Matthew	
J. Barton (Consultant)	M. Mitchell	S. Arora	M. Stroud
Maitri Banerjee (DFO)	R. Schaaf	B. Lehman	E. Sanchez
NRC Staff Presenters	K. Chang	E. Davidson	D. Lach
P. Buckberg, NRR	J. Medoff	N. Iqbal	D. Mannai
G. Meyer, RI	D. Harrison	C. Sydnor	H. Metell
J. Davis, NRR	D. Roth	J. Kim	R. Ahrabli
Entergy Presenters	M. Heath	E. Smith	T. Ivy
R. Smith	G. Cheruvenki	B. Rogers	M. Fallin
S. Bethay	D. Nguyen	A. Black	J. Hoffman
B. Sullivan	D. Reddy	D. Jeug	C. Marks
B. Ford	V. Rodriguez	T. Le	M. Gallagher
R. Pace	R. Subbaratnam		W. Teng
T. Griesbach	K. Howard		R. Plasse

room floor. The hydraulic pressure is caused by the higher water table (about 25 feet above the bottom of the reactor building) due to the geology of the site. The licensee presented various site construction diagrams showing the relative heights of the containment base mat and the ground water level. The licensee has committed to monitor the water, the concrete and the bolts and perform repair of the existing bolt corrosion.

Upon Dr. Maynard's questions, Mr. Bethay indicated that their assessment was based on monitoring of water in the torus room, that comes in and evaporates, and multiple engineering evaluations done over the years. Mr. Bethay showed pictures of various bays in the torus room, with some showing puddles of water, the worst being not of a depth that can be pumped out. Upon questions from the members, Mr. Bethay stated that only a few of the bays have water coming in, and that a small leaching of calcium from the concrete is seen in some of the bays. Upon Dr. Armijo's question, Mr. Bethay also stated that the collection containers from the drain lines in the sand bed region are periodically inspected, and always found dry, hence the containment refueling water drains can not be the source of the torus room water. A chemical analysis of the water also proves it not to be from the refueling pool.

Professor Ulm from MIT was contracted by the licensee to perform an analysis of the base mat concrete condition and cause of the water intrusion. Prof. Ulm stated that uneven heat distribution during construction of the reactor building base mat, when the outer donut structure (torus room base mat) was poured in various sections before the thicker inner core, resulted in stress concentration at the construction joints and ensuing cracks in concrete. He stated the membrane placed under the base mat had failed also. Hence, according to the licensee's analysis, ground water is finding its way through vertical construction joints and localized cracks in concrete and reaching the surface ultimately near the torus saddle support mounting bolts where the concrete is cored out by 2 to 2 ½ feet. Prof. Ulm calculated that a hypothetical cylinder with a diameter of 1/6th of an inch through the base mat in a torus room bay would result in the amount of water that has been seeping in under the hydraulic pressure.

Dr. Wallis questioned if the ground water seeping through the concrete could corrode the anchor bolts. The licensee explained that some amount of corrosion is seen, but small levels of corrosion increases the bonding between the steel and the concrete. However, as corrosion increases some cracking of the concrete around it could happen. So, they plan to do a more detailed inspection of the bolt corrosion. Regarding the reinforcing bars inside the concrete, given the small amount of leak, it is extremely unlikely, even if the whole leak is assumed to be through one 1/6 inch cylinder, that a reinforcement bar will be degraded by this leak, according to licensee's analysis. Given the lower pH value of about 9.4, measured in the torus room water, compared to the higher pH value of 13 for concrete, the licensee opined that concrete dissolution into the seeping water is occurring very slowly, thus, not to impact the structural integrity of the base mat within the license renewal time frame of concern. The small amount of calcium deposited in the bays where water is seeping in indicates a very localized but a slow process of concrete degradation not relevant to the time scale of concern.

The licensee plans to continue monitoring the water (amount, chemistry etc); and inspect a sample of the bolts and the grout around them to verify that the leakage is coming in from the area around the bolts, check the integrity of the bolts and the grout, and take necessary corrective actions. Upon Dr. Wallis' question regarding adequacy of the sample inspection every 10 years committed by the licensee, Mr. Bethay stated that routine (operator) rounds would look at the bolts, and any changes identified will be addressed through the licensee's corrective action program.

The fourth open item involved the lack of benchmarking data to support plant specific neutron

condensate storage and diesel fuel tanks that sit on sand beds. The licensee pointed out that these tanks are subject to periodic visual inspection for corrosion inside and outside and maintenance review, in addition to the confirmatory UT. Also, these carbon steel tanks are lined inside, with a relining and UT scheduled for 2008.

Dr. Abdel-Khalik asked about potential corrosion of the containment liner below the floor level due to leak from components above the floor. The licensee explained that the concrete floor is bonded to the steel liner, and UT results at points on the junction of the floor to the liner and up to one inch below have shown no degradation. These confirmatory measurements are to be continued in the future. Dr. Abdel-Khalik was concerned about the potential corrosion of the torus wall. The licensee pointed out that they have an active program of inspecting the inside torus wall and doing necessary restoration of the coating every other outage. Although the same exact locations may not be UT measured at the next outage, the licensee stated that the visible nature of the corrosion provides confidence that the program is effective in identifying degradation. Also, the licensee compares data with another plant (Fitzpatrick) of same vintage to ensure the coating is appropriately maintained.

Mr. Perry Buckberg, the NRR license renewal project manager for Pilgrim, provided an overview of the staff's license renewal review, audit and regional inspections at the plant. The audit was performed by a team of NRC staff and contractors. The audit team confirmed the licensee's scoping and screening methodology and the adequacy of the scope of systems and structures within the license renewal program. The open item related to the security diesel was reviewed by the regional inspection and closed.

The ACRS members asked if the clarity of the regulatory guidance regarding the scope of the license renewal program was adequate. This question arose due to the licensee's addition to the scope as a result of staff review. The staff stated that the identification of equipment function and environment had a role in it, and not the clarity of guidance.

Glenn Meyer, team leader of the Region I inspection team, provided a summary of their inspection regarding the license renewal program scope and implementation of the aging management program. Upon members' questions he stated that the licensee's characterization of water in the torus room matched the finding of his inspection team, and that it is not unusual to find a small amount of moisture in the torus room. Regarding the location of the catch containers (buckets) for the containment refueling water drains, the licensee stated that the buckets are in place since 1987, but one has to crawl under the torus to find them. During refuel outages, the licensee performs a visual inspection of the buckets to confirm that they are dry, and document the results.

Mr. Meyer stated that their inspection found some incorrect boundaries regarding structural interaction between safety and non-safety system components. Once identified, the licensee resolved the issue to the team's satisfaction. The team identified that a flow switch that monitors the containment refueling water leak had failed, which was later repaired. Region I plans to do another round of inspections before the period of extended operation to ensure the issues and licensee commitments are addressed. Mr. Meyer ended his presentation with a discussion on the licensee's current performance and that it was in the licensee response column. Mr. Meyer also stated that he believed that the licensee's system engineers at Pilgrim are knowledgeable and provides a generally effective program of identifying problems in the field and addressing them.

Dr. Bonaca asked if the GALL report is forcing the use of request for exceptions from the licensees and ensuing staff evaluations that could be avoided if GALL were not so prescriptive.

Dr. Wallis noted that he would have expected the licensee to clarify the torus room water issue in their application for license renewal, as it is a long standing issue, and not wait for an NRC inspection to identify it. Chairman Maynard and Mr. Barton noted that it is not uncommon to find water in some areas of the plant like the torus room (writer's note: they were probably hinting at condensation from the air due to lower area temperature and infrequent but potential small pipe leaks). Dr. Davis of the staff pointed out that the interim staff guidance on containment shell degradation requires an inspection of the torus room for water, thus making the staff more sensitive to the issue of water intrusion. Ms. Lund from NRR management noted the value of a license renewal inspection.

Chairman Maynard adjourned the meeting by thanking everyone attending the meeting.



Pilgrim Nuclear Power Station License Renewal Safety Evaluation Report

Staff Presentation to the ACRS

Perry Buckberg, Project Manager
Office of Nuclear Reactor Regulation

April 4, 2007

Introduction



- Overview
- Section 2: Scoping and Screening Review
- License Renewal Inspections
- Section 3: Aging Management Review Results
- Section 4: Time-Limited Aging Analyses (TLAAs)

Overview



-
- LRA Submitted by Letter, January 27, 2006
 - GE BWR3 - MARK 1 Containment
 - 2028 MWth, 690 MWe
 - Op License DPR-35 Expires June 8, 2012
 - Located in Plymouth, MA

Overview



- SER Issued March 1, 2007
- Four (4) Open Items
- No Confirmatory Items
- Three (3) License Conditions
- 91 RAIs Issued, 329 Audit Questions
- ≈82% Consistent With Draft GALL Report, Revision 1
- Minor Components Brought Into Scope

Review Highlights



-
- **AMP GALL Audit**
 - May 22, 2006
 - **Scoping and Screening Methodology Audit**
 - June 6 - June 9, 2006
 - **AMR GALL Audit**
 - June 19, 2006
 - **AMP/AMR Status Briefing**
 - July 17 - 19, 2006
 - **Regional Inspections**
 - September 18 - 22, 2006
 - October 2 - 6, 2006
 - December 6 - 7, 2006

Section 2: Scoping and Screening Review



Section 2.1 - Scoping and Screening Methodology

- On-site Audit - June 6 – June 9, 2006
- Staff Audit And Review Concluded That The Applicant's Methodology Satisfies The Rule (10 CFR 54.4(a) and 10 CFR 54.21)

Section 2.2 – Plant-Level Scoping

- No Omission Of Systems Or Structures Within The Scope Of License Renewal

Section 2: Scoping and Screening Review



Section 2.3 – Mechanical Systems

- 60 Mechanical Systems
- 100% Reviewed
- On-site Review Of Mechanical Systems
- Review of the Security Diesel System was Referred to the Regional Inspection Team
- Minor Components Brought Into Scope

Security Diesel



-
- Open Item 2.3.3.6: Security Diesel
 - LRA Did not Include System Drawings
 - Referred to Regional Inspector to Determine System Components in Scope
 - Staff Considers the 3/9/2007 Inspector Input Adequate to Close the Open Item

Section 2: Scoping and Screening Review



Section 2.3 – Mechanical Systems

- Components Brought Into Scope
 - Turbocharger (Emergency Diesel)
 - Turbocharger (Station Blackout Diesel)
 - Diesel Fuel Oil Emergency Transfer Skid
 - Outdoor Transformer Fire Suppression Systems

Section 2: Scoping and Screening Review



Section 2.4 – Containment, Structures, and Supports

- No Omission Of Structures Or Supports Within The Scope Of License Renewal

Section 2.5 – Electrical and Instrumentation & Control

- No Omission Of Electrical And Instrumentation & Control Systems Components Within The Scope Of License Renewal

Section 2: Scoping and Screening Summary



-
- The Applicant's Scoping Methodology Meets The Requirements Of 10 CFR Part 54
 - Scoping And Screening Results, As Amended, Included All SSCs Within The Scope Of License Renewal And Subject To AMR



License Renewal Inspections

Glenn Meyer

Region I

Scoping and Screening



-
- 54.4(a)(2) - Non-safety SSCs Whose Failure Could Impact Safety SSCs
 - Spatial and Structural Interactions
 - LRA Drawings and Procedures Reviewed
 - Plant Walkdowns Performed

Scoping and Screening Conclusions



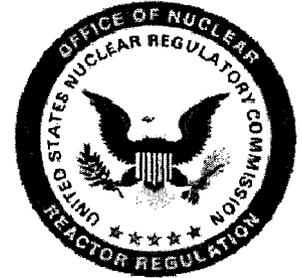
- Spatial Interaction - Acceptable
- Structural Interaction – Corrected
- Scoping and Screening Acceptable for License Renewal

Aging Management



- Reviewed 26 AMP Programs
- Reviewed Programs, Evaluations, and Records
 - Program Procedures
 - Operational Experience Information
 - Prior Pilgrim Issues
- Performed Plant Walk Downs
- Interviewed Cognizant Personnel

Inspection Conclusions



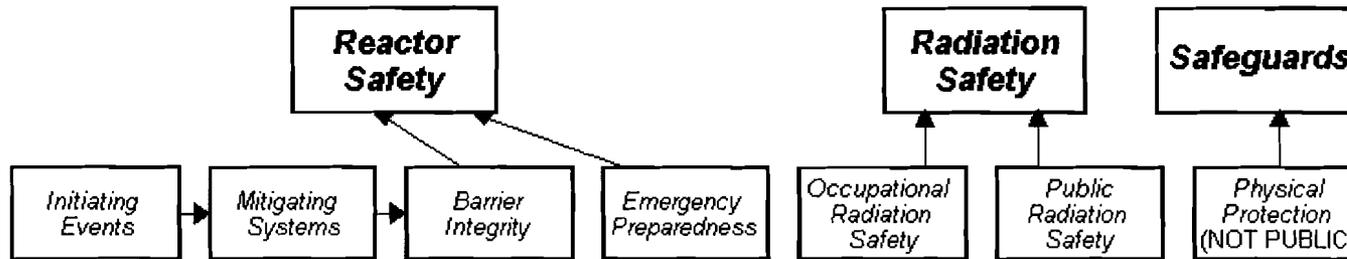
- Scoping and Aging Management Programs Support Conclusion That Aging Effects will be Managed
- Drywell Shell Monitoring – SER OI 3.0.3.3.2
- Instrumentation Circuit Testing Program

Current Performance



- Licensee Response Column (Column I) of the NRC's Action Matrix – Green PIs and Findings
- No Cross-cutting Issues
- Reactor Oversight Process Baseline Inspections

Performance Indicators

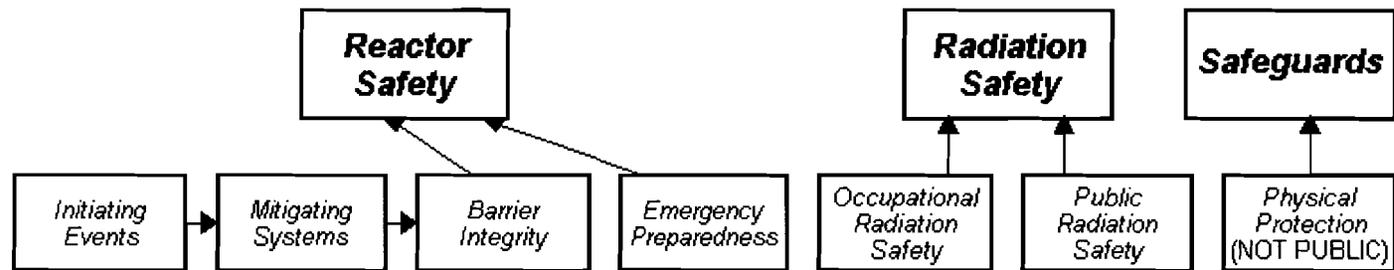


Performance Indicators

[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]
[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]		
[REDACTED]	[REDACTED]		[REDACTED]		
	[REDACTED]				
	[REDACTED]				
	[REDACTED]				



Inspection Findings



Most Significant Inspection Findings

	Initiating Events	Mitigating Systems	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety	Physical Protection (NOT PUBLIC)
4Q/2006	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]
3Q/2006	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]
2Q/2006	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]
1Q/2006	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]	[REDACTED]

Miscellaneous findings



Pilgrim Nuclear Power Station Safety Audit

Dr. James A. Davis
Audit Team Leader

Aging Management Program Audit



- 40 AMPs in the LRA Reviewed by Staff
 - 14 Programs Consistent with GALL
 - 19 Programs Consistent with GALL with Exceptions or Enhancements
 - 7 Programs are Plant Specific
 - 30 Programs are Existing Programs
 - 10 are New Programs

AMP Audit (Continued)



-
- 165 Questions During AMP Audit
 - All Questions Were Resolved

Aging Management Review Audit



- 164 Questions Closed
 - Includes One Converted To RAI (Flaw Evaluation)

Commitments



- 36 Commitments at the End of the Audit
- 8 Additional Commitments After Regional Inspection and Result of RAIs

Fire Protection Program (B.1.13.1)



-
- Open Item 3.0.3.2.10:
 - Applicant did not Adequately Address how to Manage the Aging Effects of Inaccessible Seals.

Containment Inservice Inspection Program (B.1.16.1)



- Open Item 3.0.3.3.2:
 - Regional Inspection Documented:
 - Inoperative Bellows Rupture Drain Flow Switch
 - Drain Monitoring Inconclusive & Undocumented
 - Water on Torus Room Floor

Containment Inservice Inspection Program (B.1.16.1)



- Open Item 3.0.3.3.2:
 - Applicant Actions
 - Replace Flow Switches
 - Drywell UT Data
 - Performance Evaluation of Torus Basemat
 - Verify Condition of Torus Hold Down Bolts and Grout

Section 4: Time-Limited Aging Analyses (TLAA)



- 4.1 Identification of TLAA
- 4.2 Reactor Vessel Neutron Embrittlement
- 4.3 Metal Fatigue
- 4.4 Environmental Qualification Analyses of Electrical Equipment
- 4.5 Concrete Containment Tendon Prestress
- 4.6 Containment Liner Plate, Metal Containment, and Penetrations Fatigue Analysis
- 4.7 Other TLAA
 - 4.7.1 Reflood Thermal Shock of the Reactor Vessel Internals
 - 4.7.2 TLAA in BWRVIP Documents

Section 4.2: Reactor Vessel Neutron Embrittlement



-
- Six TLAA's Affected by Neutron Fluence
 - Reactor Vessel Fluence
 - Pressure-Temperature Limits
 - Upper Shelf Energy
 - Adjusted Reference Temperature
 - Circumferential Weld Inspection Relief
 - Axial Weld Failure Probability
 - TLAA Analyses was Submitted and Deemed Not Acceptable

Neutron Fluence



- Open Item 4.2

- RAMA Methodology Used to Calculate Neutron Fluence
- Dosimetry Data was not Available with Which to Benchmark the Calculated Results.
- Result - Fluence Calculation Not Acceptable Per Reg Guide 1.190 -Lead to Open Item 4.2

Neutron Fluence



-
- Open Item 4.2
 - Applicant's Submittal IAW Described Plan
 - Applicant Completes Fluence Evaluation and Verifies TLAA Basis
 - The Staff Will Review Submittal for Acceptance

Section 4.3: Metal Fatigue



-
- Commitments 31 and 35 Will Ensure That Either
 - Projected 60 yrs Cycles Enveloped by Design Cycles
 - Refined CUF ≤ 1 for PEO
 - Aging Effects Will be Managed for the Components
 - Repair Or Replace the Affected RPV Locations
 - Based on These Commitments, the Staff Accepted the Evaluations in accordance with 10 CFR 54.21(c)(1)(i),(ii) and (iii)



Section 4.4: Environmental Qualification (EQ) of Electrical Equipment

- Applicant's EQ Program consistent with GALL AMP X.E1, "Environmental Qualification of Electrical Equipment"
- Staff Concluded The EQ Program Is Adequate To Manage The Effects Of Aging On The Intended Function Of Electrical Components
- The Staff Accepted the Evaluation in Accordance with 10 CFR 54.21(c)(1)(iii)

Conclusions

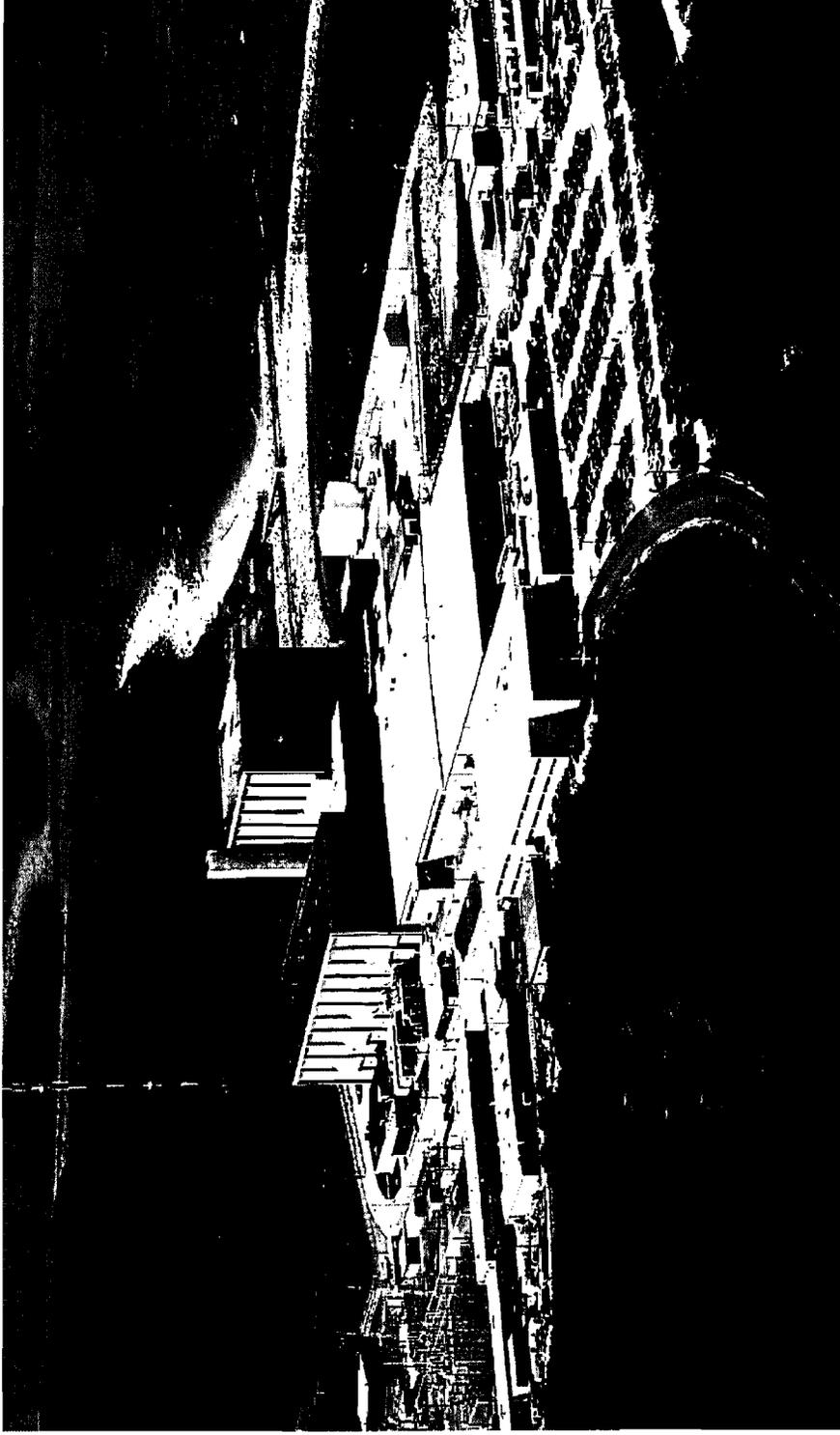


- On the basis of its review of the LRA, with the exception of Open Item (OI) 4.2, and pending resolution of OIs 2.3.3.6, 3.0.3.2.10 and 3.0.3.3.2, the staff determines that the requirements of 10 CFR 54.29(a) have been met.

Pilgrim Nuclear Power Station

ACRS License Renewal Subcommittee Presentation

April 4, 2007



Pilgrim Personnel in Attendance

Bob Smith	General Manager Plant Operations
Steve Bethay	Director of Nuclear Safety Assurance
Brian Sullivan	Director of Engineering
Bryan Ford	Licensing Manager
Ray Pace	Mechanical Engineering Design Supervisor
Alan Cox	Entergy LR Project Manager
Fred Mogolesko	Pilgrim LR Project Manager
Other support personnel	

Agenda

- Pilgrim Description and Current Status
- Pilgrim Licensing History/Highlights
- License Renewal Project
- Cost-Beneficial SAMAs
- Pilgrim SER Open Items
 - Security Diesel Generator
 - Fire Barrier Penetration Seals
 - Containment Inservice Inspection
 - Neutron Fluence
- Questions

Pilgrim Description

- Located in Plymouth, Massachusetts on Cape Cod Bay
- ~ 40 miles south of Boston
- Sited on 1600 Acres
- BWR-3
- Mark I Containment
- General Electric (NSSS), Bechtel (AE and Constructor)
- 2028 MWt Thermal Power; ~ 690 MWe
- Open Cycle Condenser Cooling
- Owned and Operated by Entergy
- Staff: ~ 655

Current Plant Status

- Coastdown began February 1, 2007
- 2007 RFO-16 begins April 6, 2007
- NRC PIs & Inspection Findings All Column 1

Pilgrim Licensing History/Highlights

- Construction Permit August 26, 1968
- Operating License June 8, 1972
- Full Power License September 15, 1972
- Commercial Operation December 9, 1972
- License Transfer to Entergy July 13, 1999
- Appendix K Power Uprate (1.5%) May 8, 2003
- LR Application Submitted January 25, 2006
- Operating License Expires June 8, 2012

Pilgrim Licensing History/Highlights

(continued)

- 1977- Replaced Core Spray safe-ends and piping inside primary containment with IGSCC-resistant material
- 1978 -1982 Mark I containment modifications
- 1984 - Replaced recirculation piping to address IGSCC concerns
- 1986 -1989 Safety enhancement modifications (SSW-RHR cross-tie, Direct Torus Vent to Main Stack, Station Blackout Diesel Generator)
- 1991 - Hydrogen water chemistry
- 1995 - Replaced ECCS suction strainers

Pilgrim License Renewal Project

- LRA Prepared by experienced, multi-discipline Entergy team (corporate and on-site)
- Pilgrim and VY LRAs first applications submitted following issuance of Rev. 1, SRP and GALL
- Incorporated lessons learned from previous applications
- Peer review conducted (10 Utilities)
- LRA internal reviews (OSRC, SRC, QA)

Pilgrim License Renewal Project

(continued)

- Commitments in the LRA refined as needed during audit/inspection process (40 aging management programs)
- Commitments tracked - Pilgrim commitment tracking system
- 40 aging management programs
 - 14 programs in place w/o enhancements
 - 16 programs require enhancement
 - 10 New Programs

Aging Management Program Implementation

Cost-Beneficial SAMAs

- Evaluating Cost-Beneficial SAMAs

SAMA 30	Procedures to cross-tie AC buses
SAMA 34	Procedures to cross-tie DC buses
SAMA 53	Containment venting within narrow pressure band
SAMA 56	Redundant DC power to DTV valves
SAMA 57	Use of diesel fire pump fuel transfer pump
SAMA 58	Power SR Division A load center bus loads via NSR Bus, SR Division B loads via NSR Bus, when SR 4kV unavailable post-trip
New	Portable power source to SR 125 vdc Battery

- None are age-related

- SAMA Details

Pilgrim Draft SER Open Items

- OI 2.3.3.6 Security Diesel Generator
- OI 3.0.3.2.10 Fire Barrier Penetration Seals
- OI 3.0.3.3.2 Containment Inservice Inspection
- OI 4.2 Reactor Vessel Neutron Fluence

Security Diesel Generator

OI 2.3.3.6

- Region 1 Confirmatory Item
 - Determine whether security diesel components are within the scope of license renewal
- Requested Support Provided

Fire Barrier Penetration Seals

OI 3.0.3.2.10

- Concern on aging management of inaccessible seals
- All penetration seals are included in the inspection program

Containment Inservice Inspection

OI 3.0.3.3.2

- Potential for corrosion in the inaccessible area of the steel containment shell, base mat and sand pocket region
- Inspection Observations
 - Rupture drain flow switch
 - Surveillance documentation
 - Torus room floor water

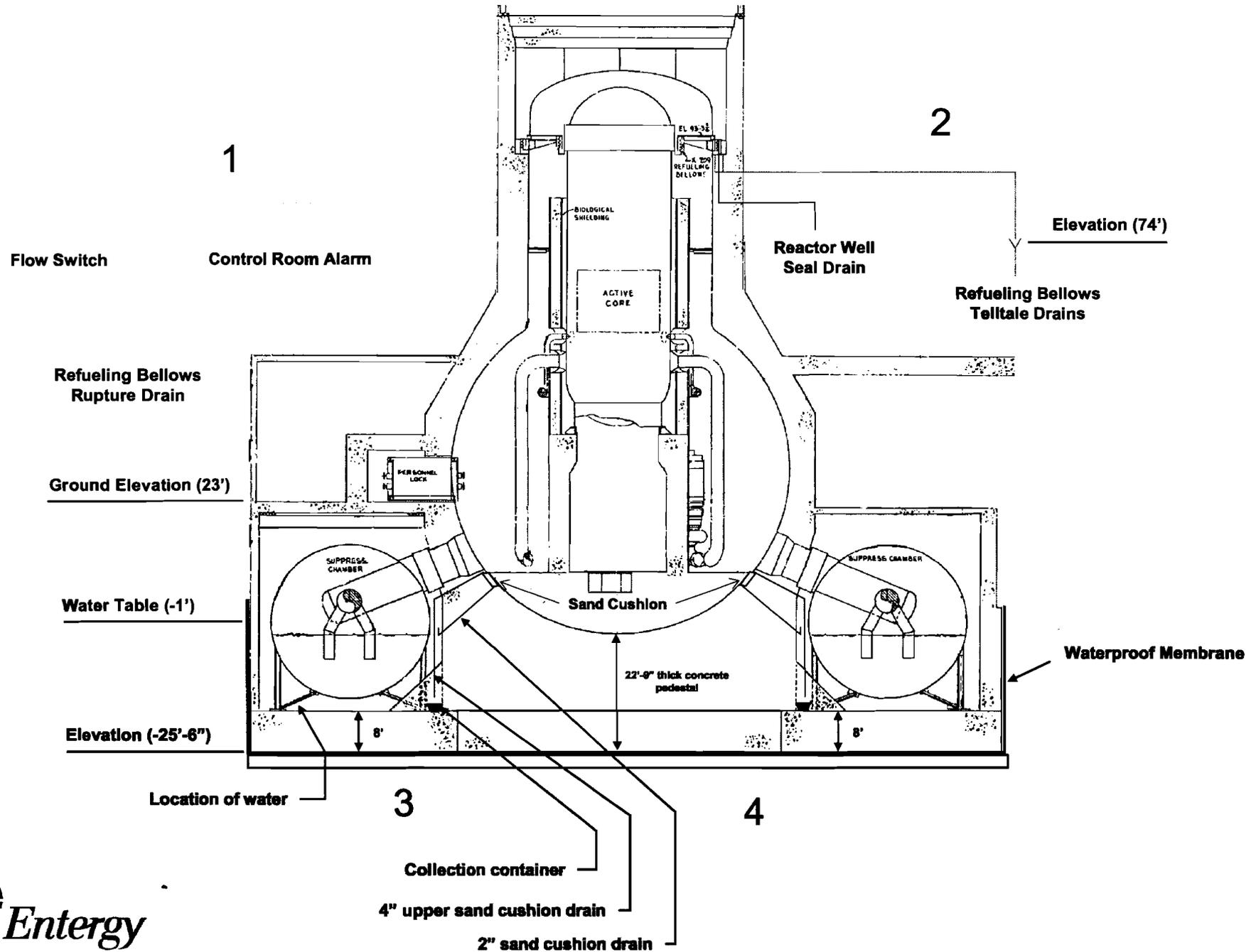
Containment Inservice Inspection

Drywell Shell Condition and Monitoring

- Design minimizes potential for undetected water intrusion
- Diverse methods of prevention and identification of potential water leakage into air gap
- No Refueling Bellows leakage
- No water intrusion into drywell air gap
- No drywell shell degradation
- Confirmatory inspections performed and planned

Containment Inservice Inspection

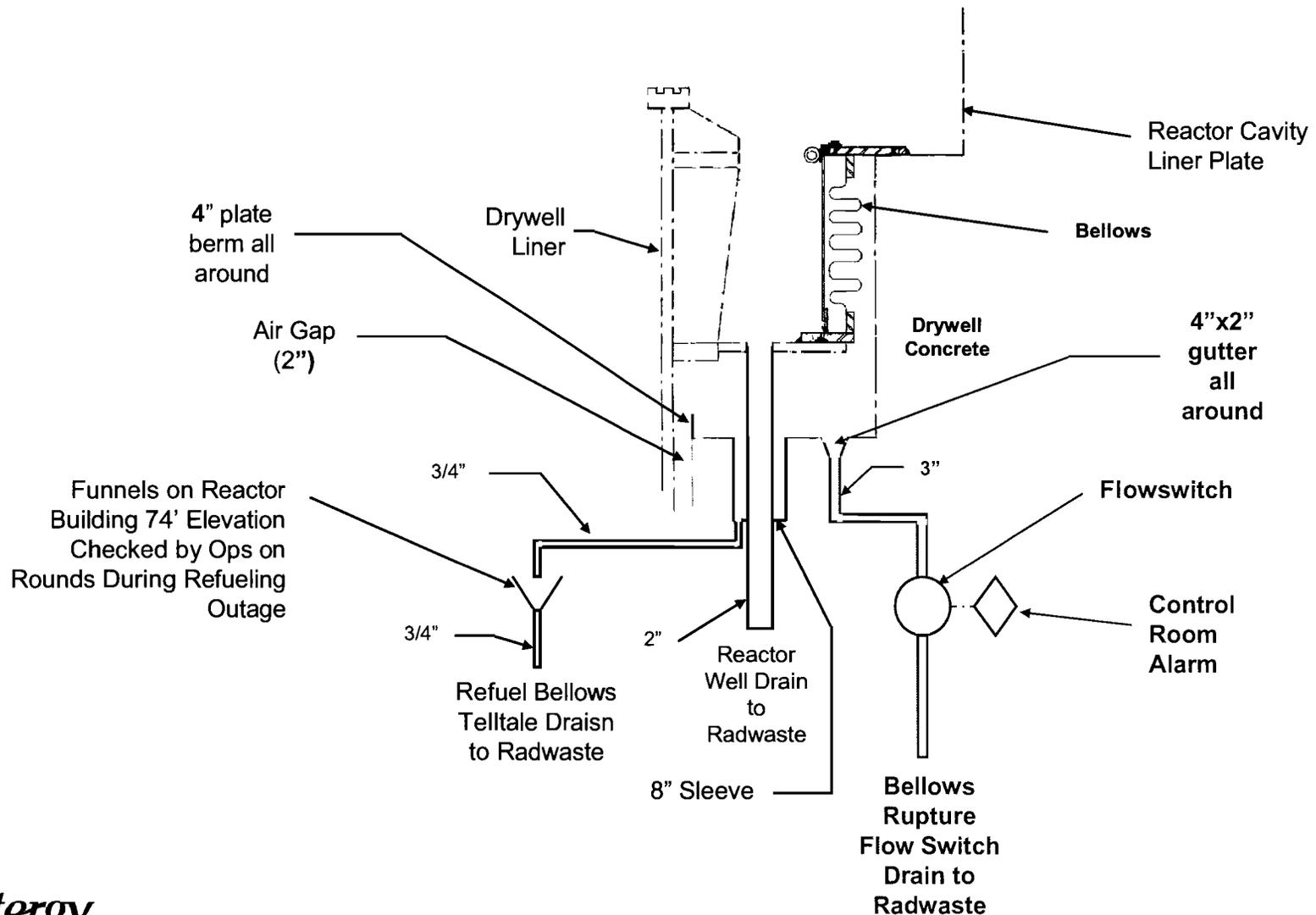
Drywell Shell Condition and Monitoring



Containment Inservice Inspection

Drywell Shell Condition and Monitoring

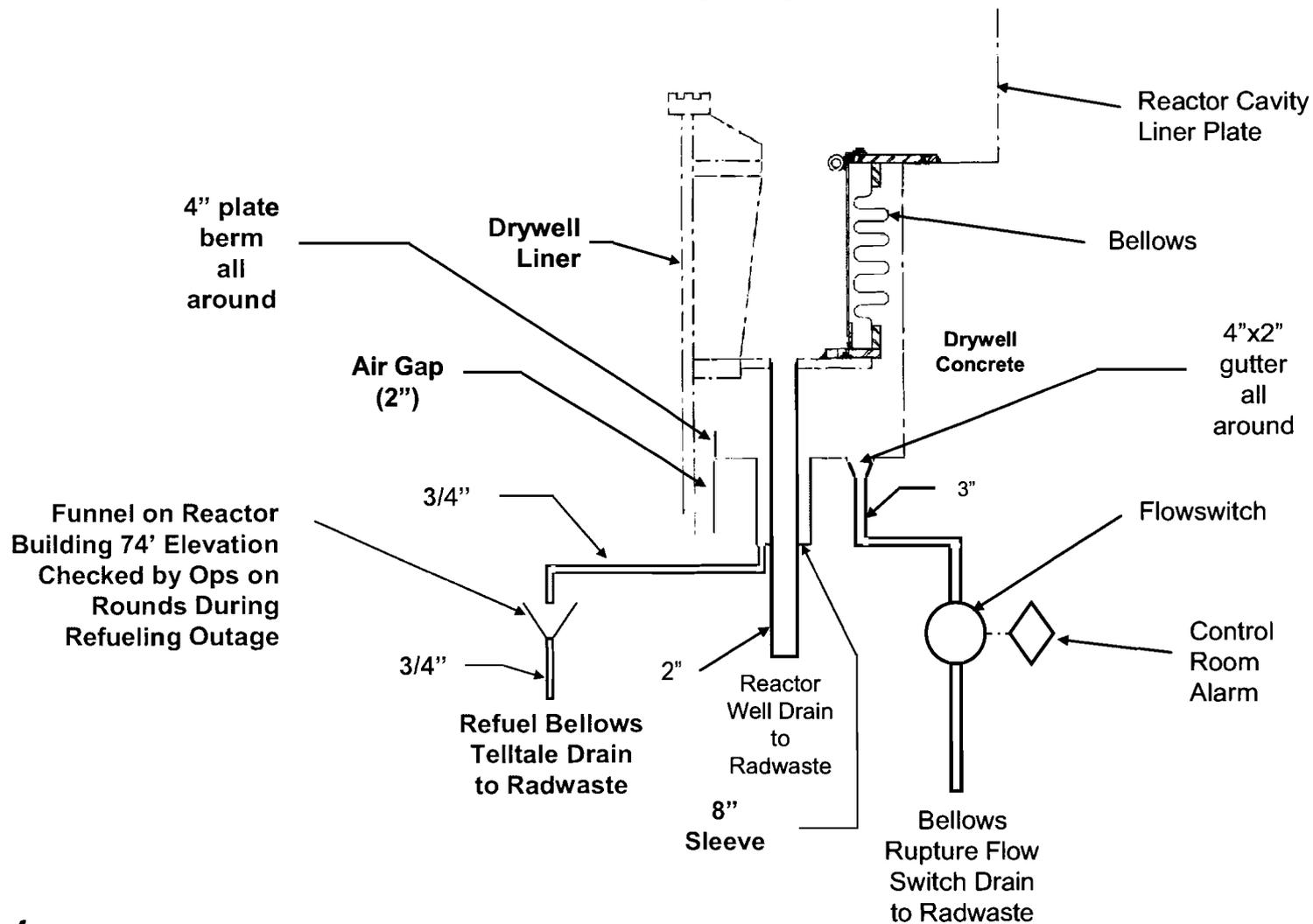
3" instrumented drain line alarms in control room



Containment Inservice Inspection

Drywell Shell Condition and Monitoring

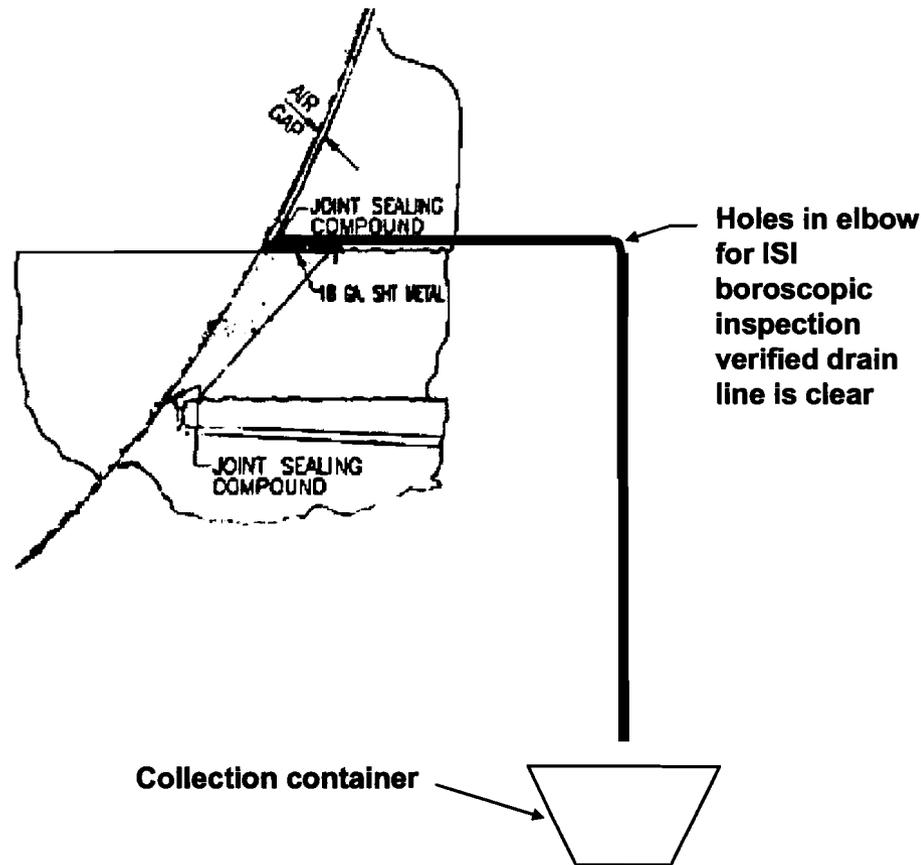
Four $\frac{3}{4}$ " drain lines which exit to 74' floor drains checked during operator tours



Containment Inservice Inspection

Drywell Shell Condition and Monitoring

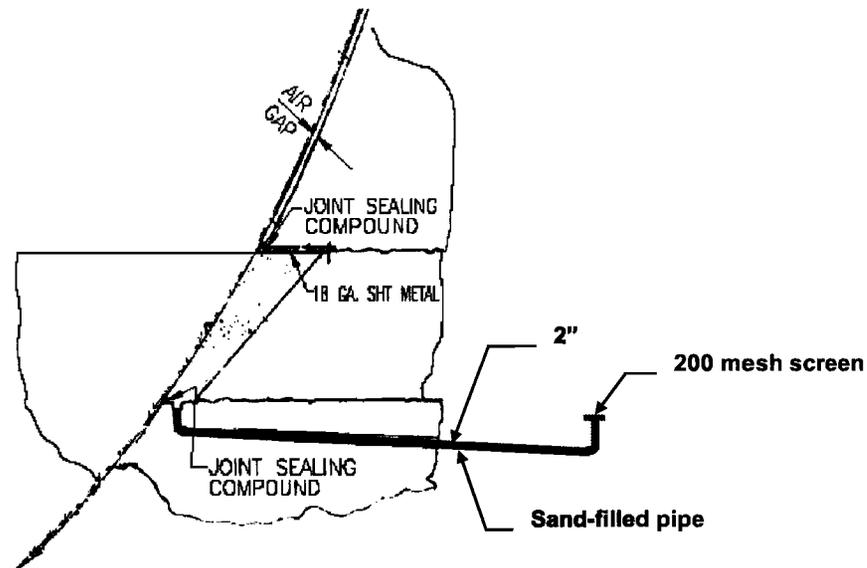
**Four 4" upper sand cushion drains
drain into collection devices and are
monitored at beginning and end of each RFO**



Containment Inservice Inspection

Drywell Shell Condition and Monitoring

Four sand cushion drains provide further detection capabilities



Containment Inservice Inspection

Drywell Shell Condition and Monitoring

Past Inspections

- Limited confirmatory UT
 - UT at twelve locations at 9'-2" elevation
 - UT at four locations at 9'-1" elevation
 - Concrete chipped out to a depth of 1"
 - UT at six locations at 72' and 83' elevations
- Verified upper sand cushion drains unobstructed and dry
- All inspections identified no corrosion

Containment Inservice Inspection

Drywell Shell Condition and Monitoring

Future Inspections

- UTs at 12 Locations at 9'-2" Elevation
 - Prior to Period of Extended Operation
 - Once within first 10 years
- UTs at 4 Locations at 9'-1" Elevation
 - Prior to Period of Extended Operation
 - Once within first 10 years
- UT at 72' Elevation Adjacent to SFP
 - Conducted every 40 months by IWE

Containment Inservice Inspection

Water on Torus Room Floor

Containment Inservice Inspection

Water on Torus Room Floor

Aspects Evaluated

- Source of water
- Integrity of concrete and anchor bolts
- Structural adequacy of the reactor building
- Inspection and monitoring of water, concrete, and Torus hold down anchor bolts

Containment Inservice Inspection

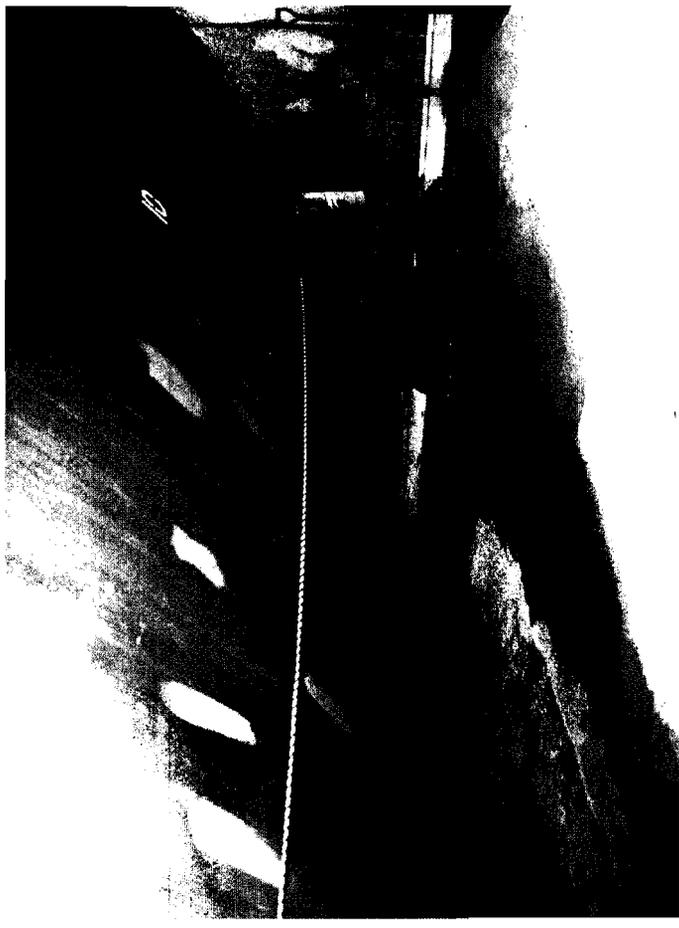
Water on Torus Room Floor

- Water on torus room floor was initially identified in late 1970's and early 1980's.
- Multiple Engineering Evaluations

Containment Inservice Inspection

Torus Room Floor

Bay 10



Containment Inservice Inspection

Water on Torus Room Floor

Evaluation

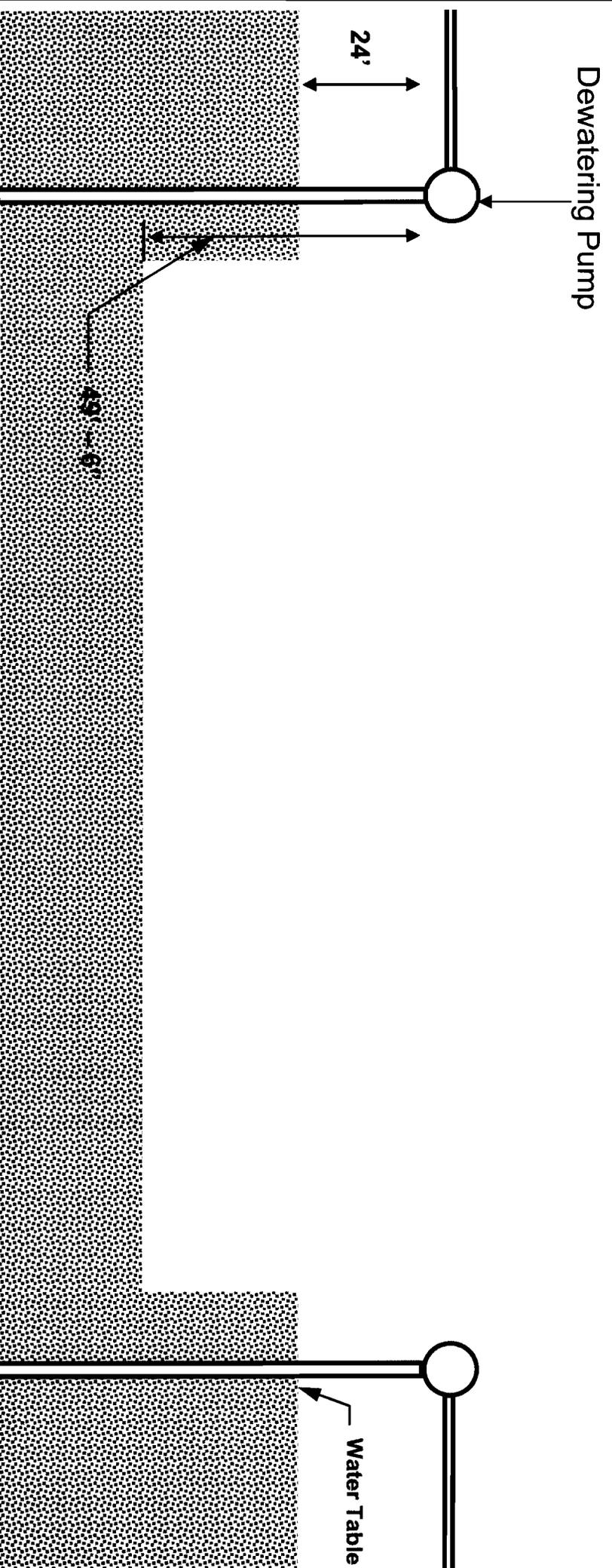
- The source is ground water seepage under hydraulic pressure.
- Low seepage rate is counteracted by evaporation.
- Non aggressive, benign water chemistry.
- No structural distress evident
 - No cracks other than hairline
 - No evidence of structural settlement
- Grout around Williams Anchor Bolts performs no structural integrity function.

Containment Inservice Inspection

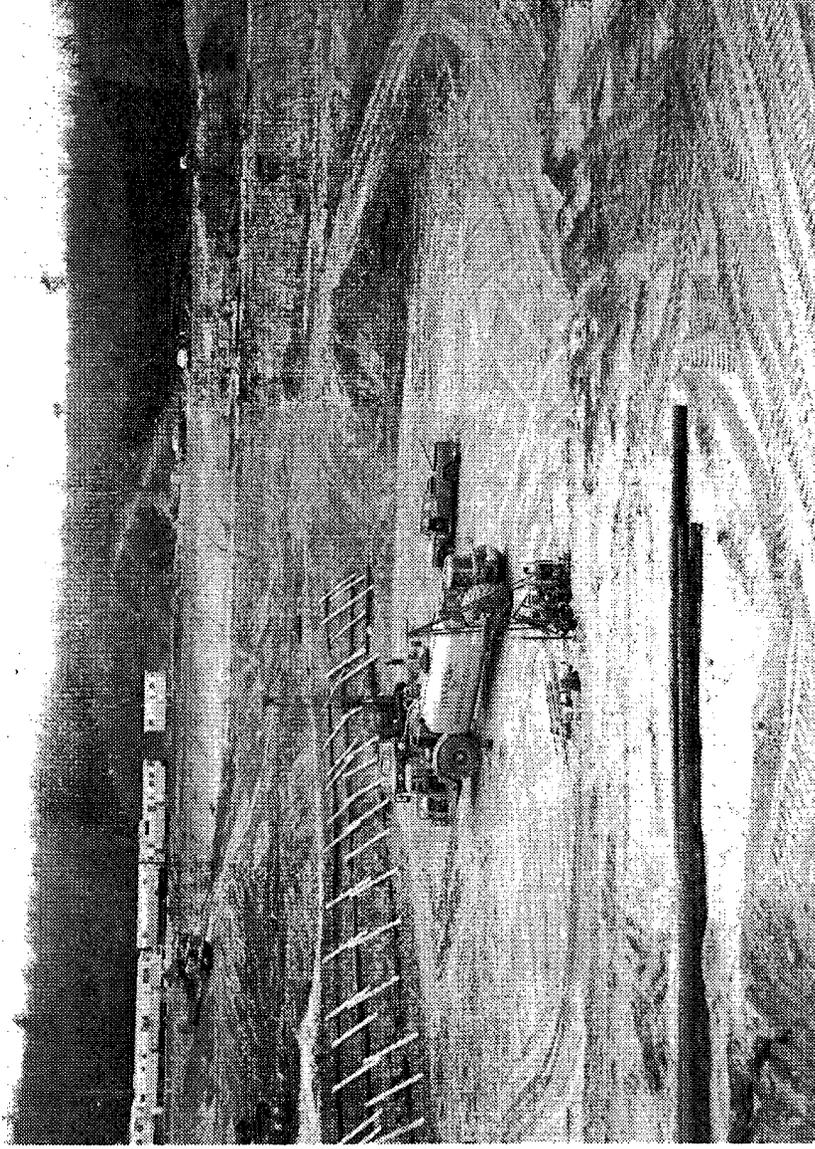
Water on Torus Room Floor

- Current Status
 - Water re-analyzed to demonstrate non aggressiveness.
 - Review over last 6-weeks shows minimal water on torus floor.
 - Seepage, mostly in Bay 10 estimated at $\frac{1}{4}$ " to $\frac{1}{2}$ " over 100 square feet.
 - Independent Assessment to demonstrate full functional capability of torus base-mat conducted.
 - Professor Franz Ulm of MIT's Department of Civil Engineering

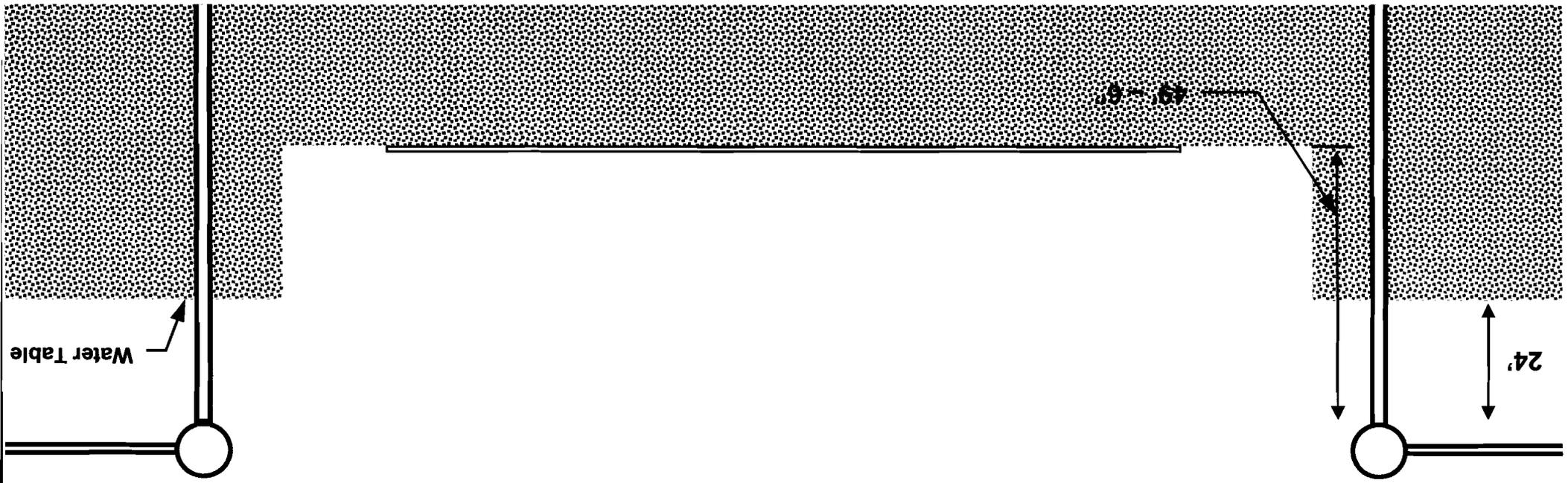
Reactor Building Site Preparation



Reactor Building Site Preparation



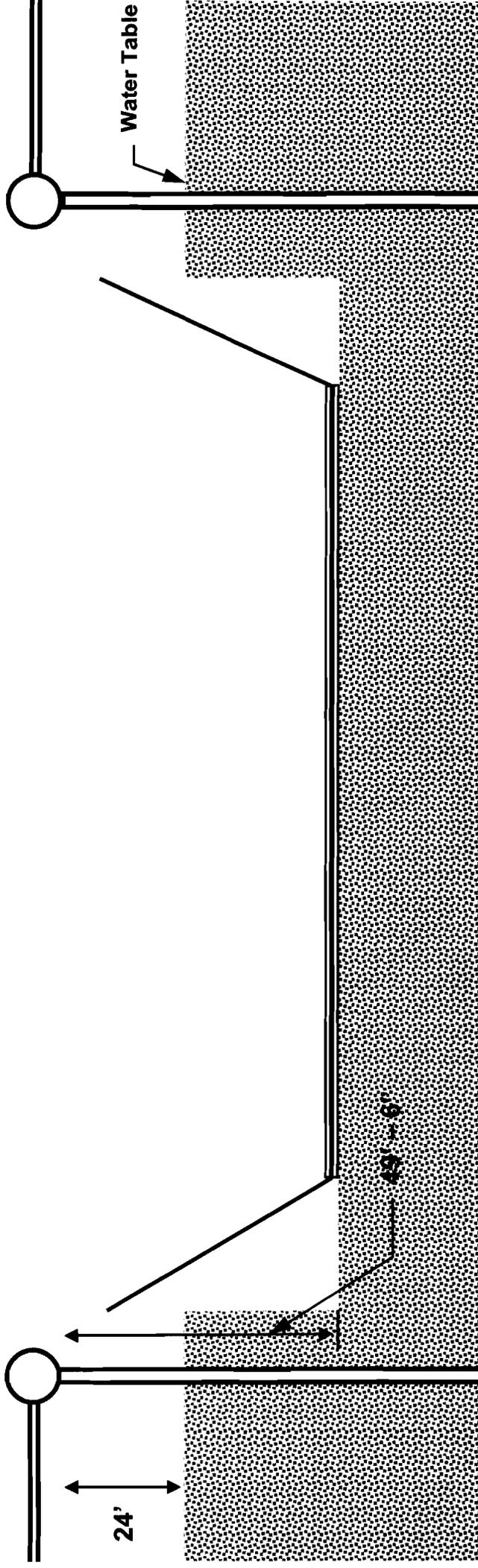
Place 3" Work Slab



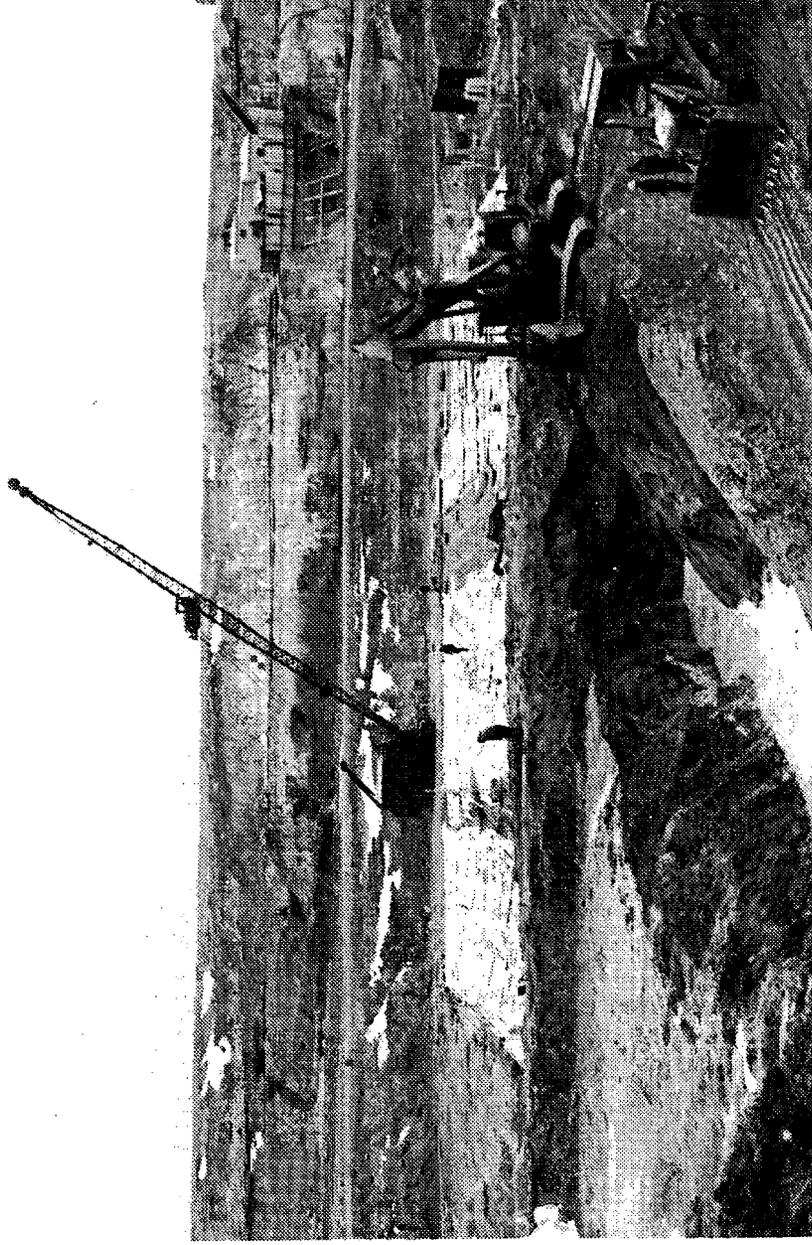
Place 3" Work Slab



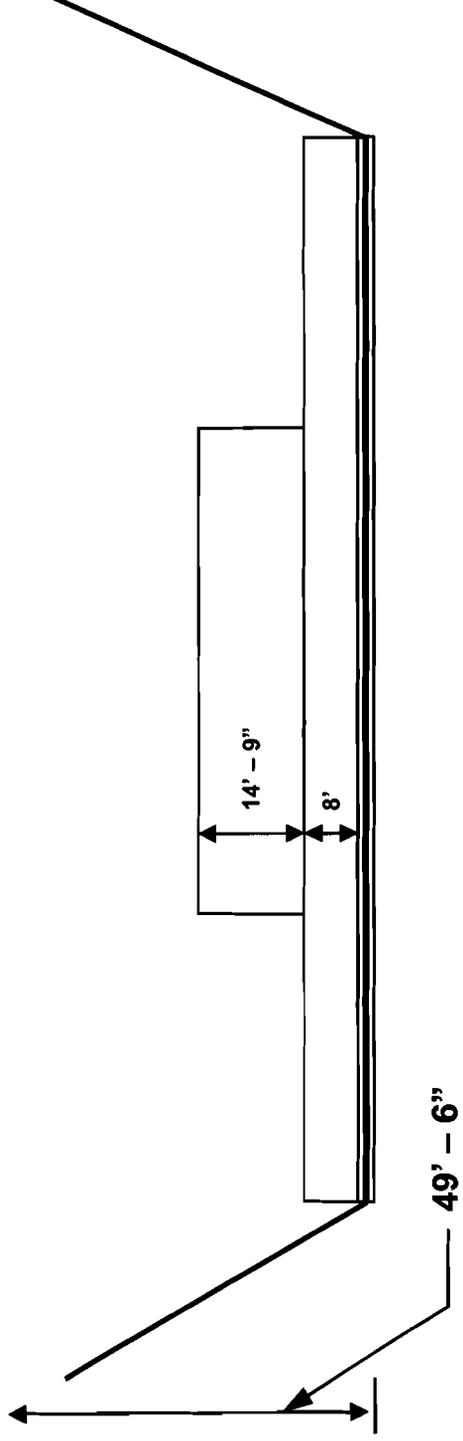
Foundation Membrane



Foundation Membrane

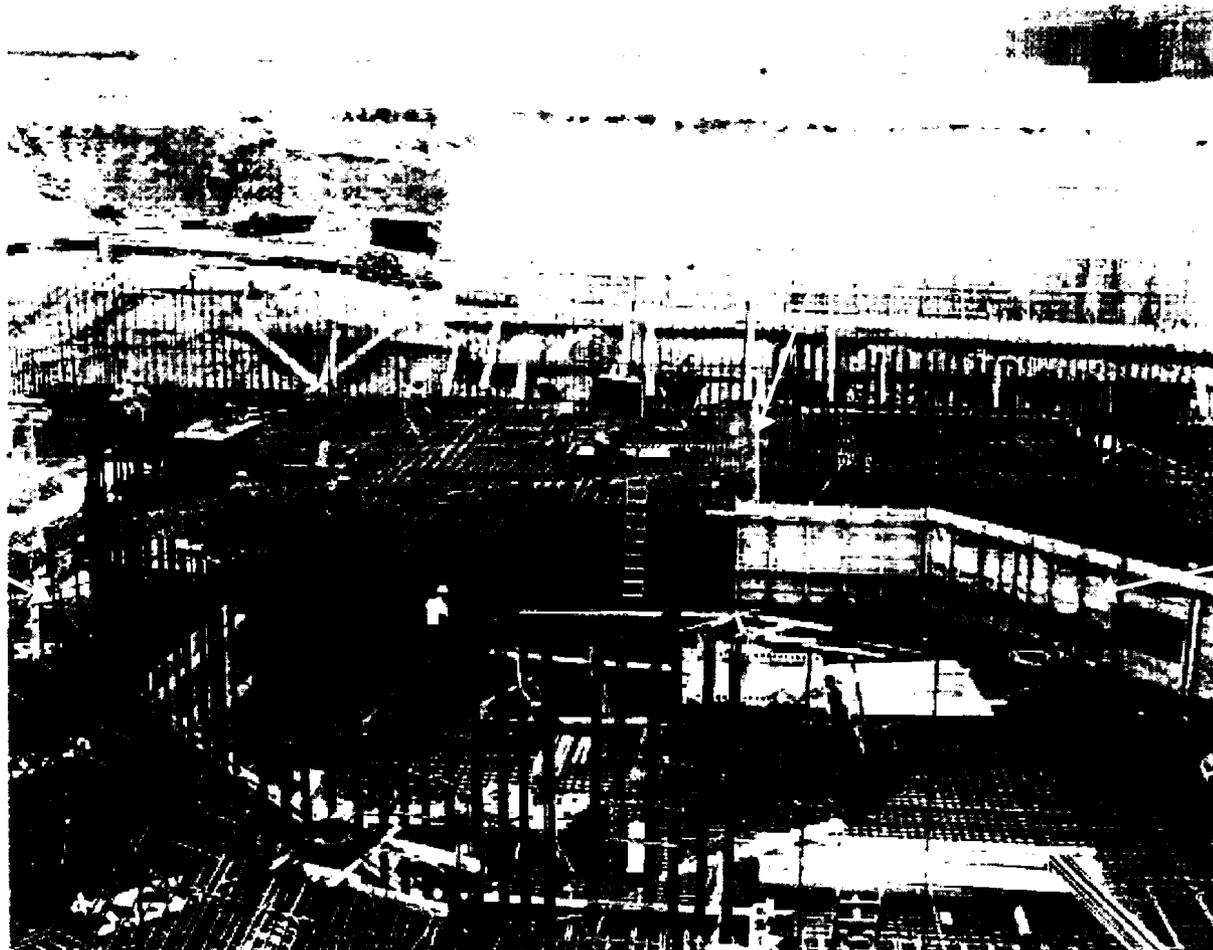


Base Mat Construction



Reactor Building Base Mat Rebar

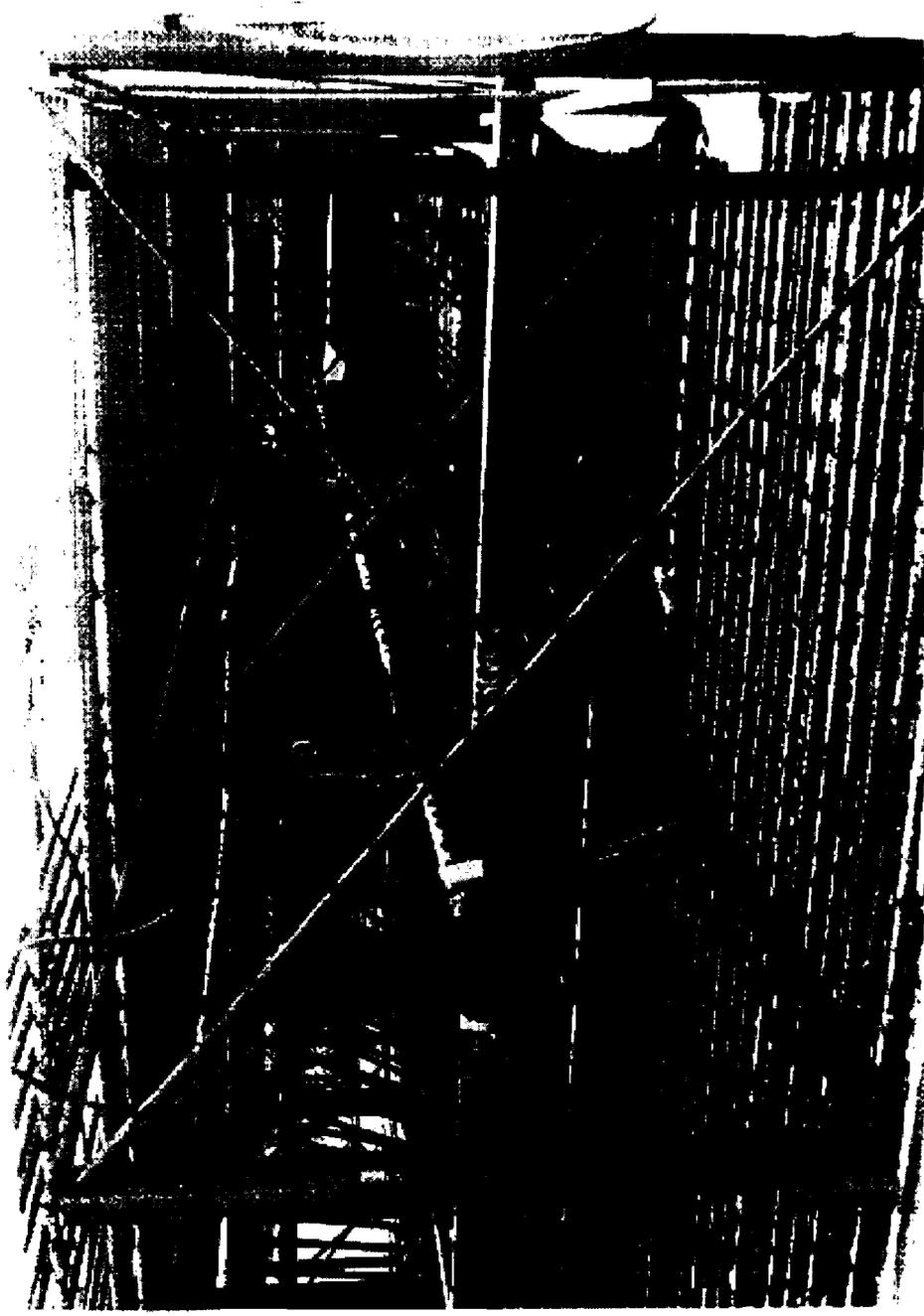
Construction Joint



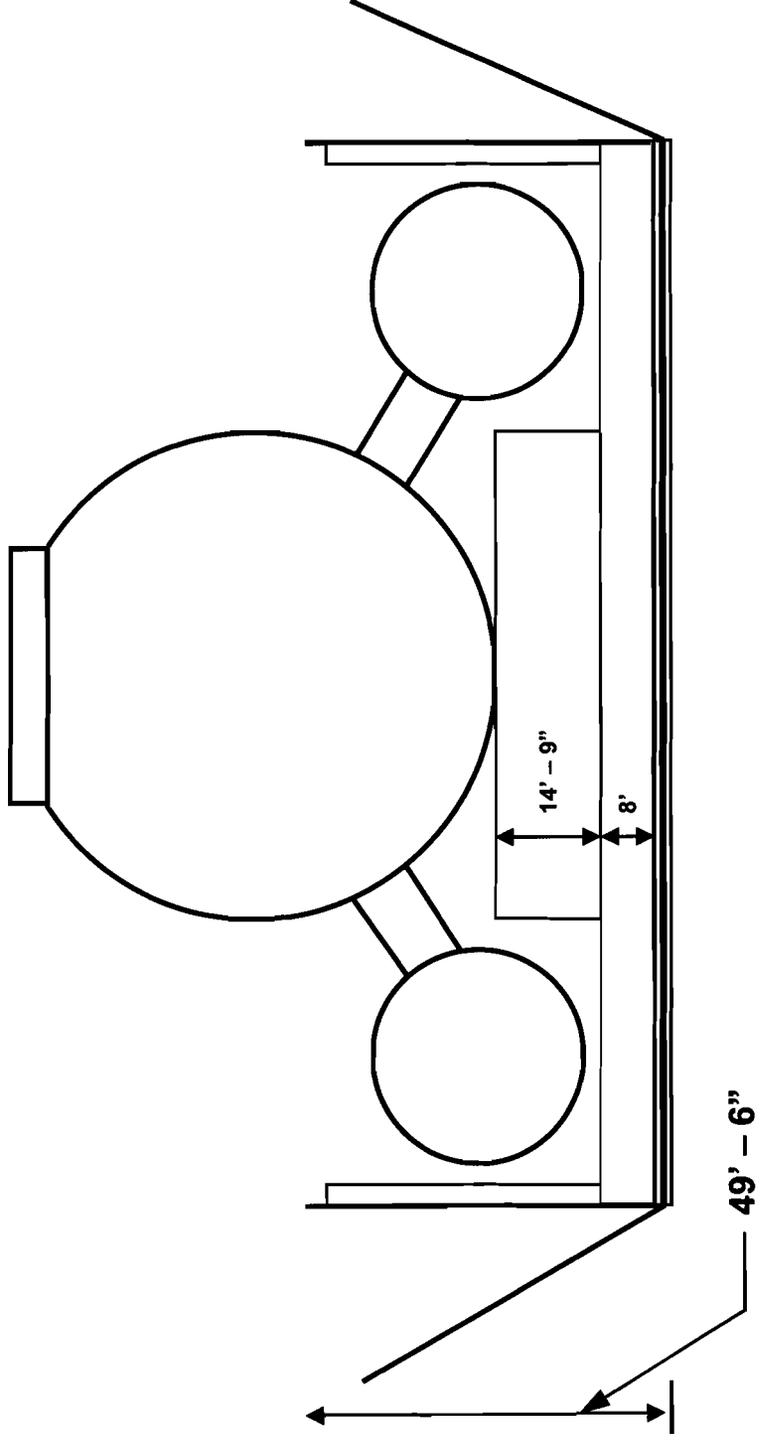
Construction
Joint

Location of
Octagonal
Construction Joint
Under Torus

Reactor Building Base Mat Rebar



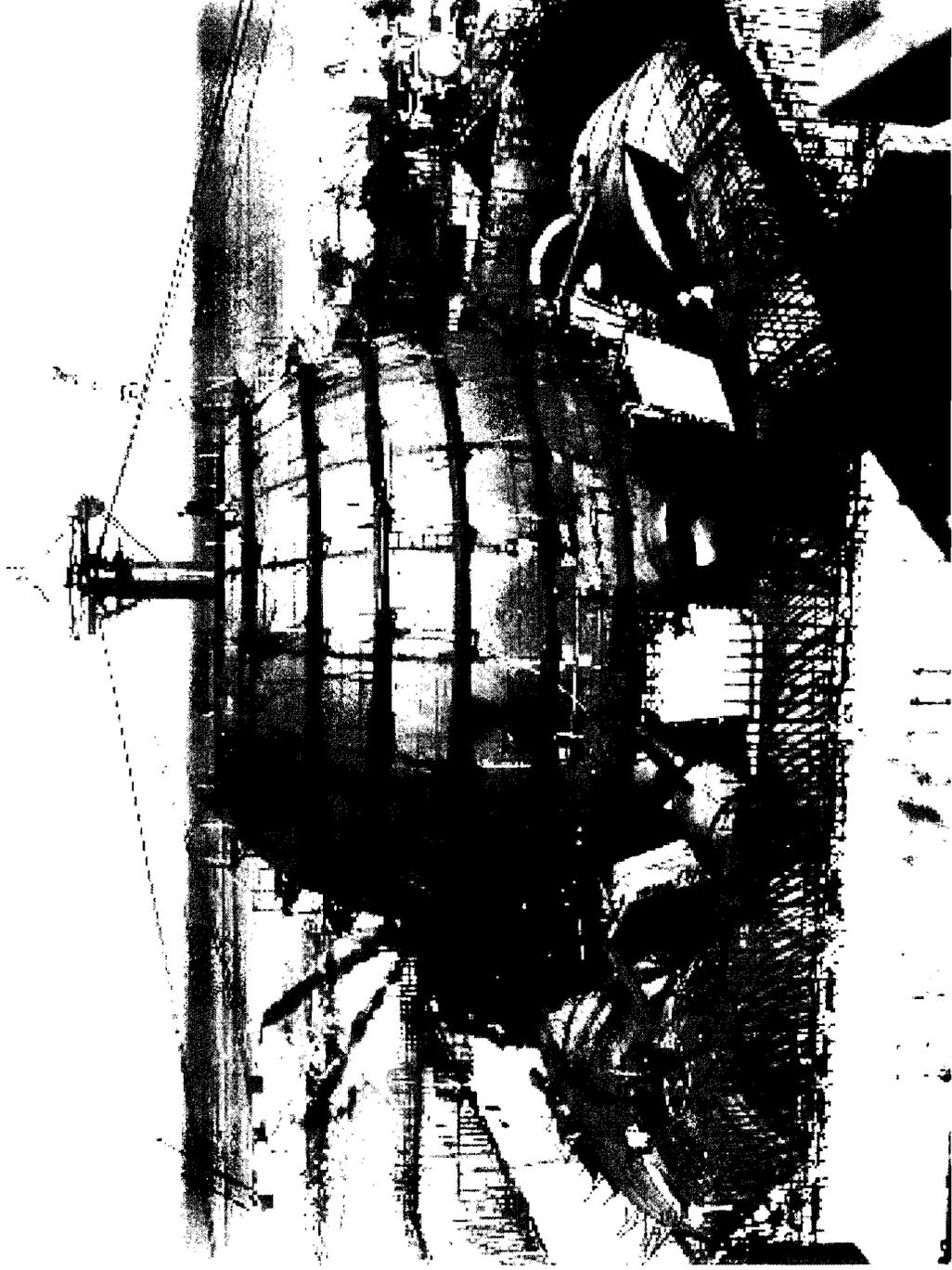
Drywell & Torus Construction



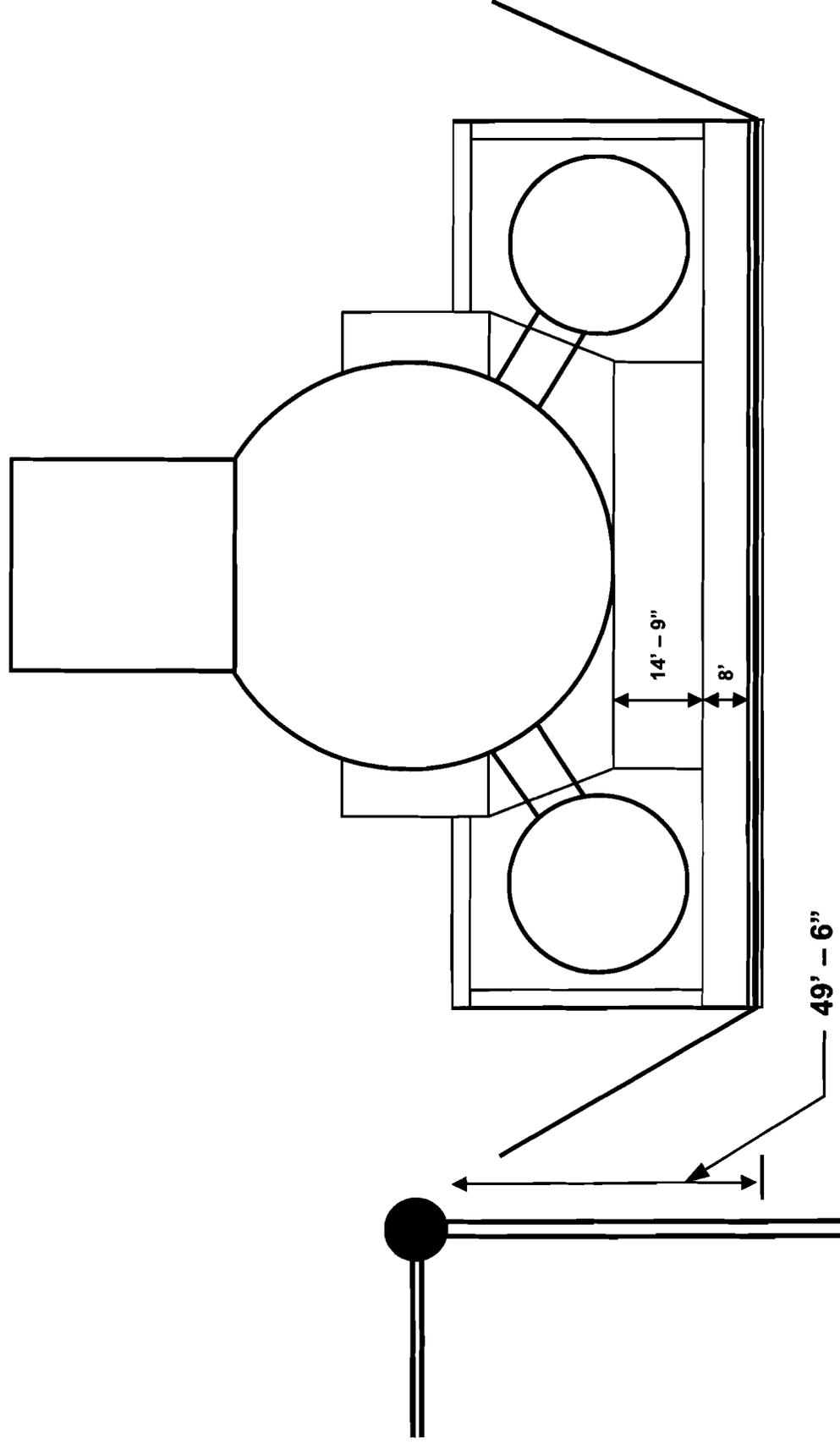
Torus Construction



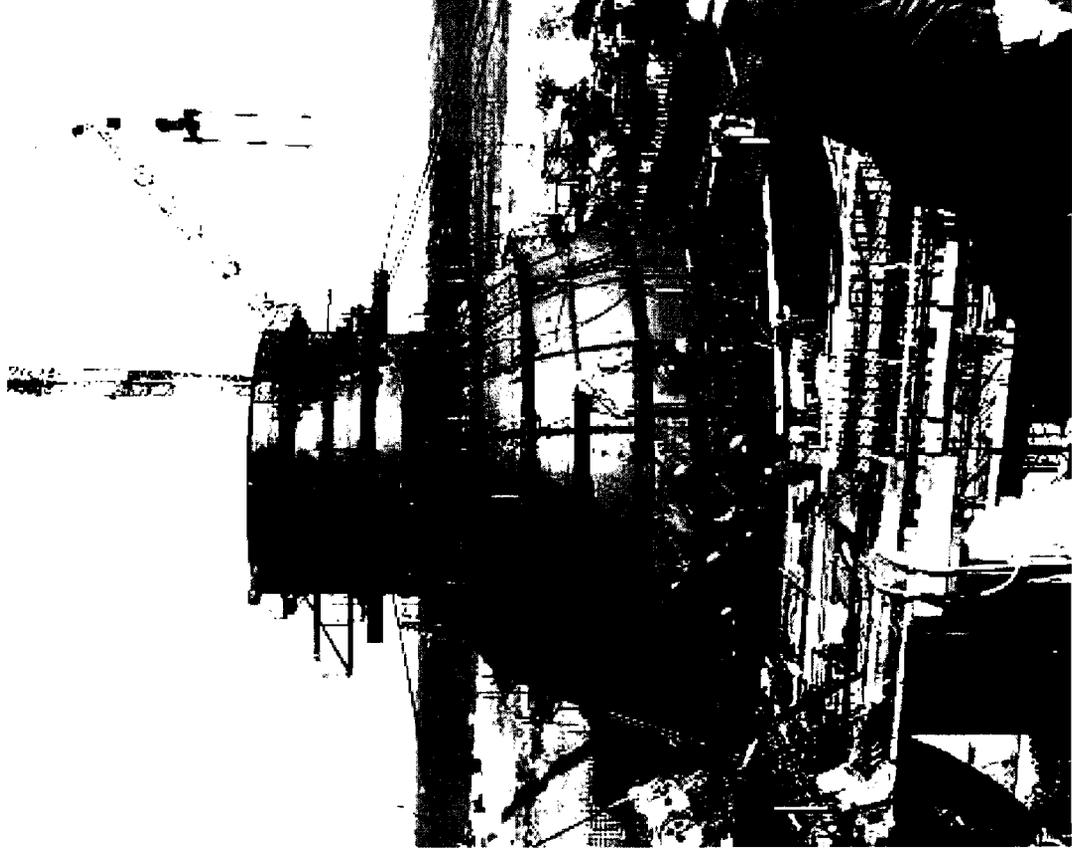
Drywell & Torus Construction



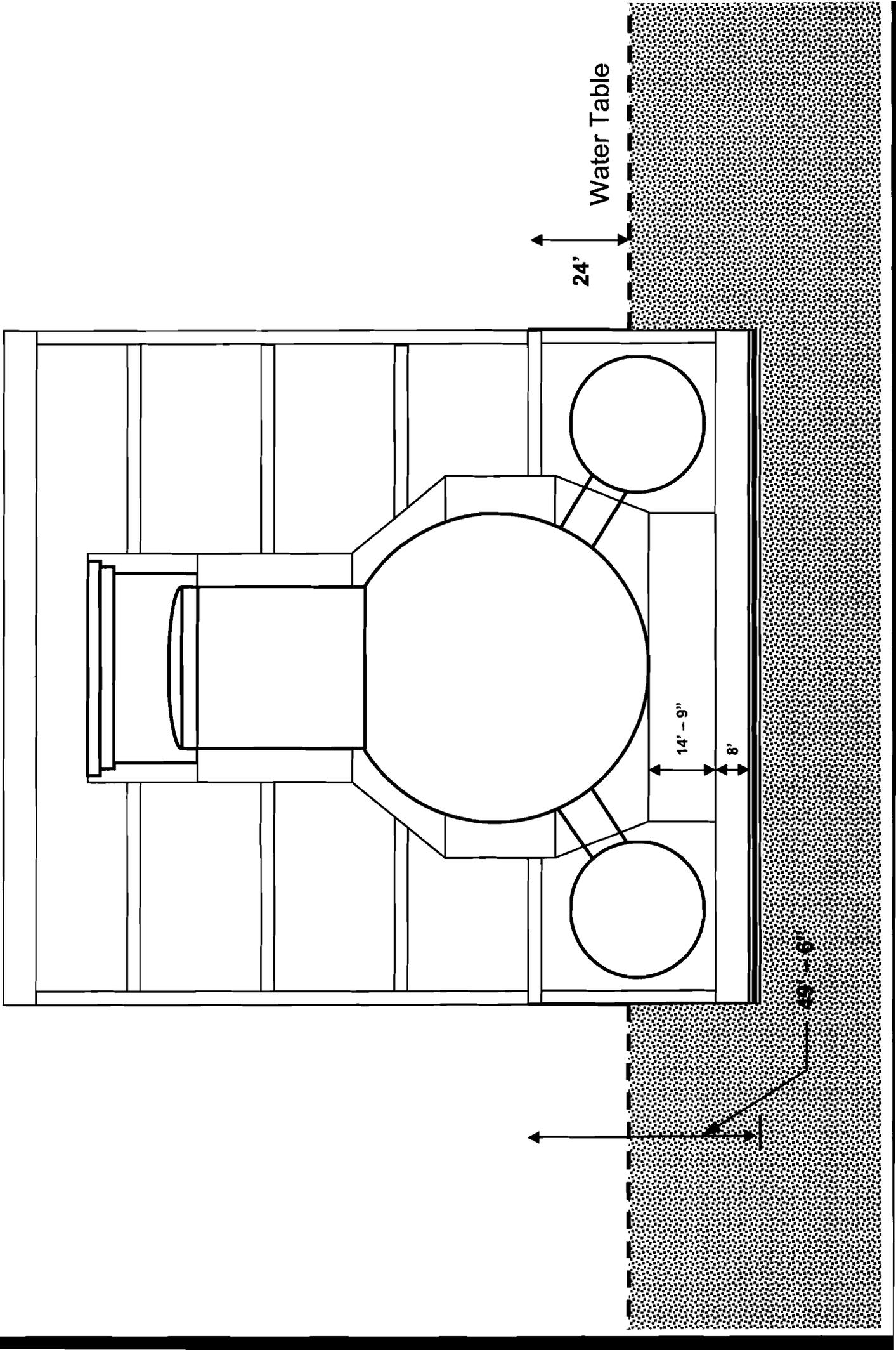
Drywell Construction



Drywell Construction

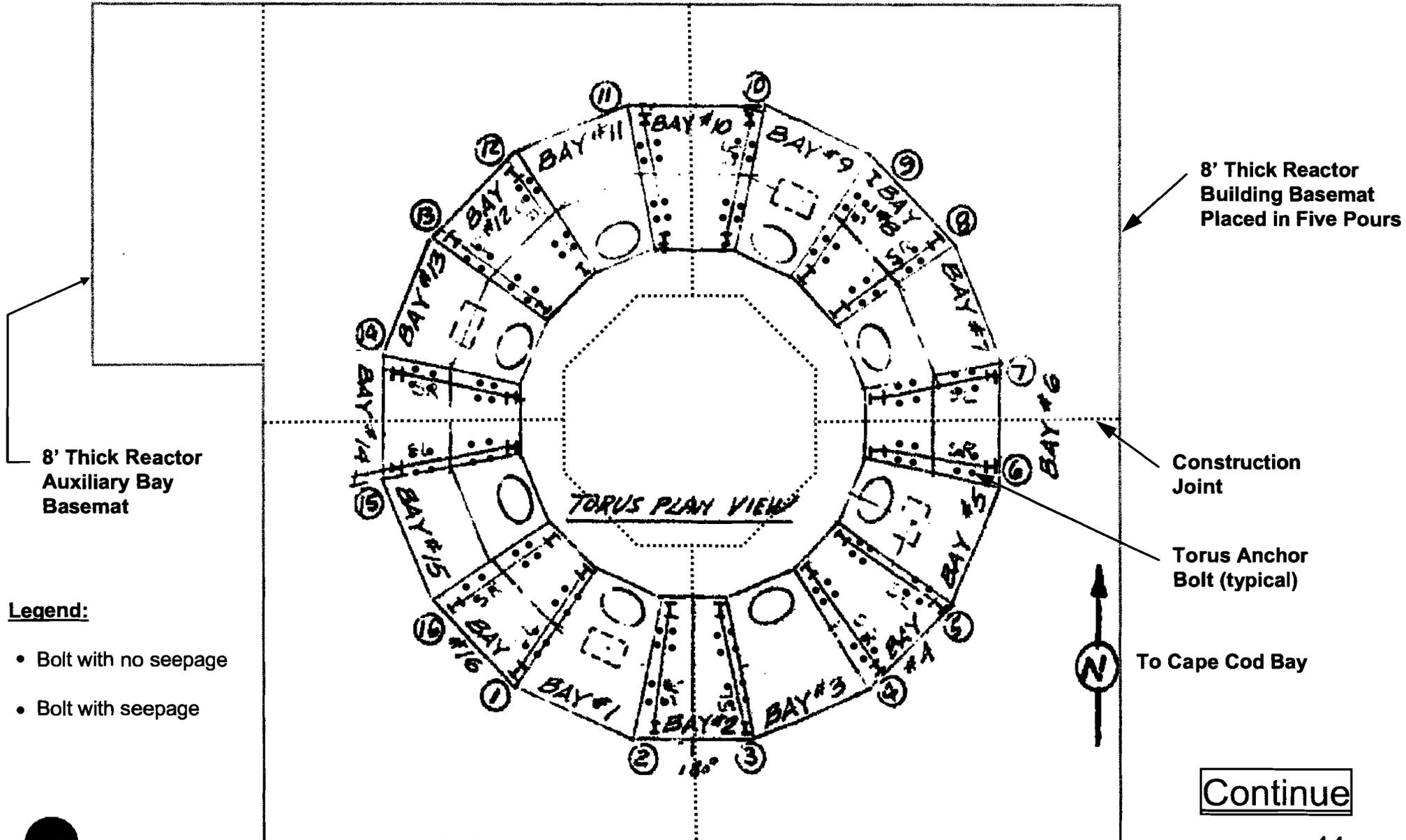


Completed Reactor Building



Containment Inservice Inspection

Torus Room Floor



Continue

Containment Inservice Inspection

Torus Room Floor

Bay 1



Back

Containment Inservice Inspection

Torus Room Floor

Bay 2



Back

Containment Inservice Inspection

Torus Room Floor

Bay 3



Back

Containment Inservice Inspection

Torus Room Floor

Bay 4



Back

Containment Inservice Inspection

Torus Room Floor

Bay 5



Back

Containment Inservice Inspection

Torus Room Floor

Bay 6



Back

Containment Inservice Inspection

Torus Room Floor

Bay 7



Back

Containment Inservice Inspection

Torus Room Floor

Bay 8



Back

Containment Inservice Inspection

Torus Room Floor

Bay 9

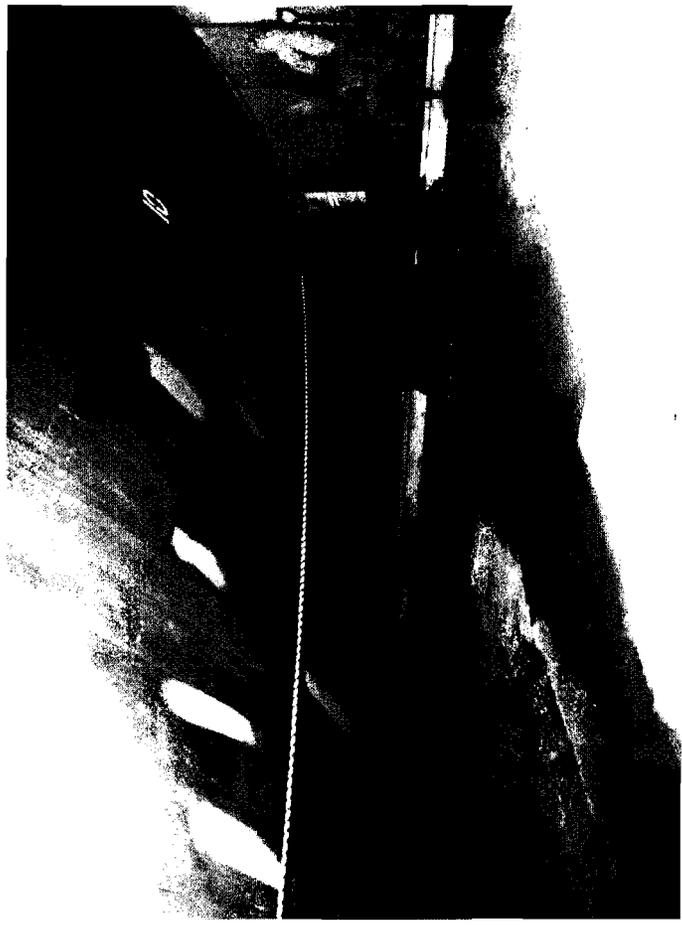
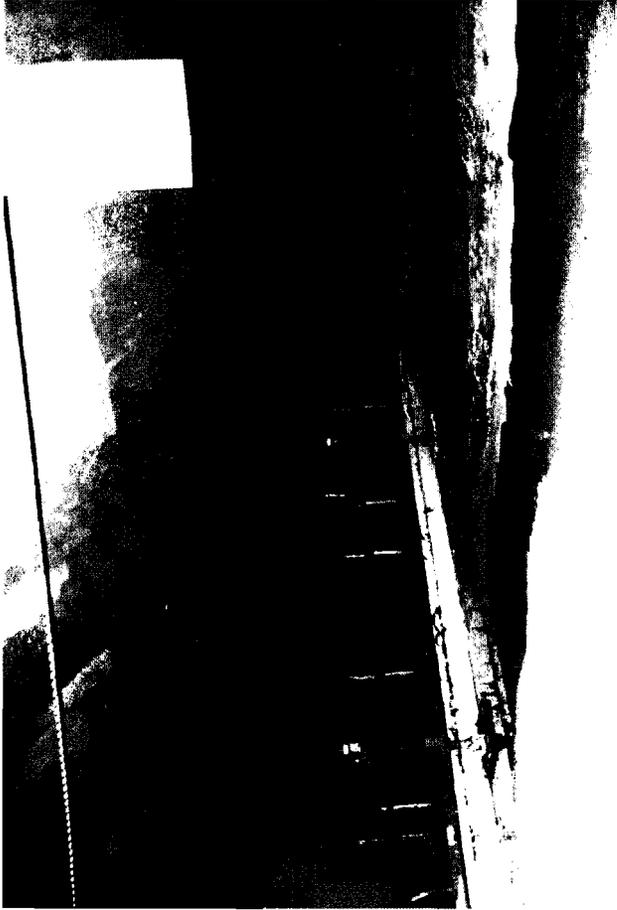


Back

Containment Inservice Inspection

Torus Room Floor

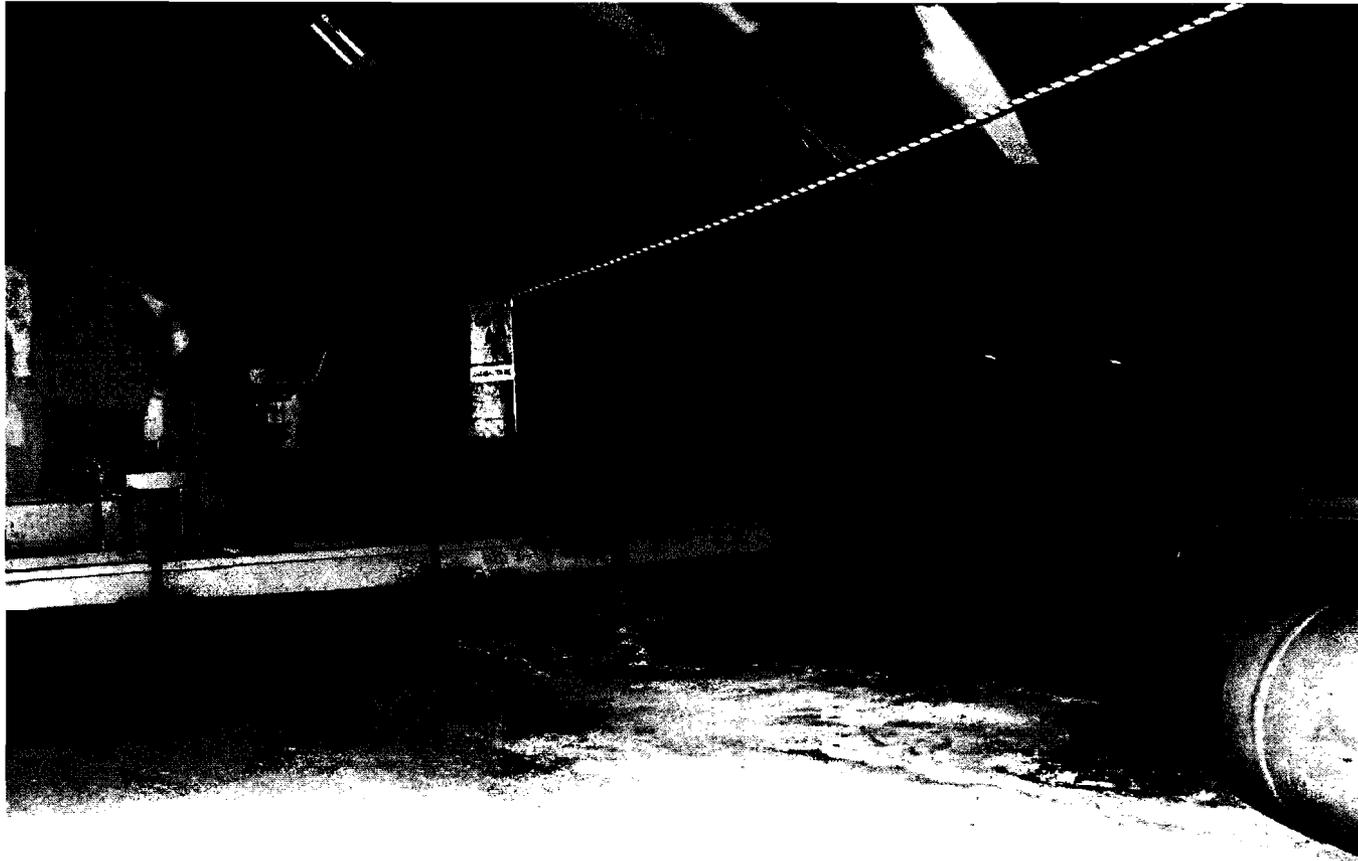
Bay 10



Containment Inservice Inspection

Torus Room Floor

Bay 11



Containment Inservice Inspection

Torus Room Floor

Bay 12



Containment Inservice Inspection

Torus Room Floor

Bay 13

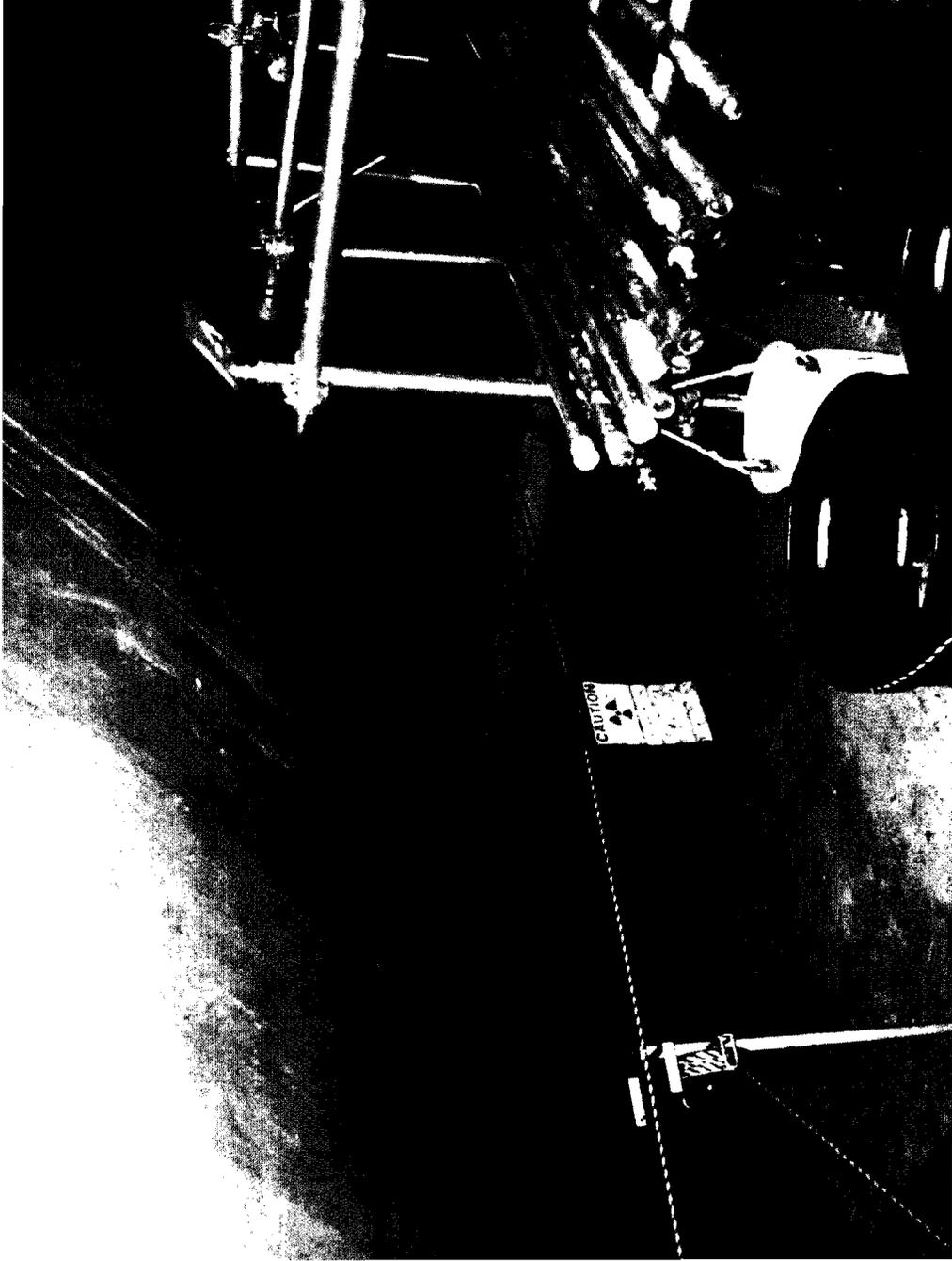


Back

Containment Inservice Inspection

Torus Room Floor

Bay 14



Back

Containment Inservice Inspection

Torus Room Floor

Bay 15

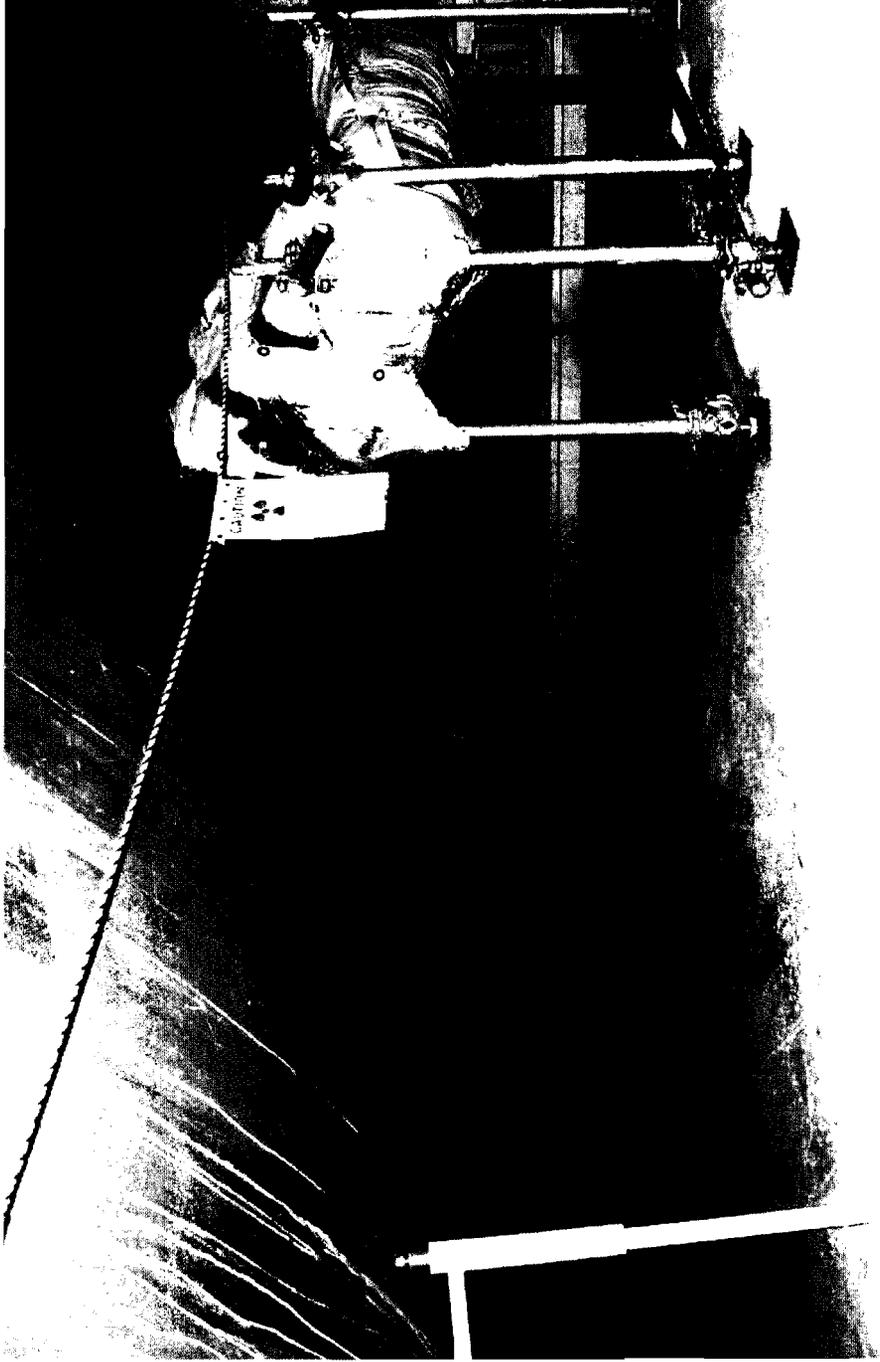


Back

Containment Inservice Inspection

Torus Room Floor

Bay 16



Back

Containment Inservice Inspection

Torus Room Floor

- **Concrete Cracking**

- Pilgrim built to ACI 318-63 concrete code.
- ACI 318 and NUREG/CR-6927 recognize concrete will crack under shrinkage, temperature and tension loading.
- Design of concrete structures assumes cracking, and reinforcing steel is added to ensure structural integrity.
- ACI 318 requires reinforcing steel to minimize cracking.

- **Leaching and Efflorescence**

- Very limited leaching and efflorescence noted.
- NUREG/CR-6927 notes this is unlikely to be an issue for high quality, low permeability concretes.

Containment Inservice Inspection

Water on Torus Room Floor

- **Concrete Water Chemistry**

- Minimum degradation threshold limits for concrete established:
 - Acidic solutions with pH < 5.5
 - Chloride solutions > 500 ppm
 - Sulfate solutions > 1500 ppm
- Pilgrim groundwater is non-aggressive to base-mat:

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

- Torus Room Floor Water Samples:

<u>Date</u>	<u>pH</u>	<u>Chlorides (ppm)</u>	<u>Sulfates (ppm)</u>	<u>Calcium (ppm)</u>
3/89	8.76	120		292
4/99	9.5			
2/07	9.45			
3/07	9.29	560	9.1	6.8

Containment Inservice Inspection

Water on Torus Room Floor

- Assessment Findings
 - Groundwater migration is highly localized.
 - Path is through vertical joints and zones most likely weakened by tensions generated during setting and hydration following the construction.
 - Localized zones are discontinuities equivalent to a vertical cylindrical hole of a maximum diameter of 4 mm (1/6th in).
 - The localized calcium leaching does not affect the overall structural performance of the slab.

Containment Inservice Inspection

Water on Torus Room Floor

- Assessment Recommendations:
 - Calcium leaching may have degraded the grout in the annular space between the 3 in. diameter hole and the 2 in. diameter Williams rock anchors. An inspection of the grout and bolt is recommended.
- Assessment Conclusion:
 - The highly localized nature of the zones through which water penetrates, does not compromise the overall structural performance of the Torus base mat. It does not affect the bulk integrity of the concrete slab or the overall compressive and bending load bearing capacity of the reactor foundation.
 - Non-aggressiveness of ground water verified.
 - Condition of Anchor Bolts to be verified.

Containment Inservice Inspection

Torus Room Floor

- Verify Condition of Torus Hold Down Bolts (by sample)
 - Prior to Period of Extended Operation
- Determine Additional Actions Based on Torus Hold Down Bolts Inspection and Water Chemistry
 - Prior to Period of Extended Operation
- Monitor Chemistry Of Groundwater
 - Every Five Years
- Monitor Chemistry of Water on Floor
 - Prior to Period of Extended Operation
 - Once in First Ten Years
- Inspect Structure in Accordance with Structures Monitoring Program
 - Every Five Years

Reactor Vessel Neutron Fluence

OI 4.2

- Lack of benchmarking data to support plant specific fluence calculations for use in TLAAAs

Reactor Vessel Neutron Fluence

- **Current Licensing Basis**
 - 10 CFR 50 App. G requirement for current operation
 - Current P-T curves valid through Cycle 18 (2011 RFO)
 - Commitment to Submit Action Plan for Resolution by September 15, 2007
 - Commitment to submit RG 1.190 calculations by June 8, 2010

Reactor Vessel Neutron Fluence

- Current Actions
 - Evaluate TLAAAs to determine limiting Fluence
 - Limiting Beltline Adjusted Reference Temperature
 - Upper Shelf Energy
 - RPV Internals
 - RPV Welds
 - RPV nozzles near beltline
 - Core Shroud Fluence Limiting based on BWRVIP-35
 - Limiting Fluence values will not be challenged

Reactor Vessel Neutron Fluence

- Future Actions
 - Benchmark Computer Code using Pilgrim or other BWR3 Dosimetry Data
 - Previous commitment to submit action plan by September 15, 2007
 - License Condition to submit calculations consistent with RG 1.190 by June 8, 2010 which demonstrate limiting fluence values will not be reached during period of extended operation.

DATE: March 26, 2007

TO: Mr. Otto Maynard, Chairman
Plant License Renewal Committee

FROM: John J. Barton
ACRS Consultant

SUBJECT: LICENSE RENEWAL APPLICATION
PILGRIM NUCLEAR POWER STATION

Having completed my review of the License Renewal Application, the NRC Safety Evaluation Report, and other Documents, I have the following questions and/or comments.

A. Pilgrim Nuclear Power Station Audit and Review Report:

1. Sect. 3.3.2.3 "AMR Results Not Consistent or Not Addressed in Gall Report." Pg. 372 - Aluminum In Air-Outdoor, Internal and External Environment
The discussion concludes that Aluminum exposed to an outdoor air environment does not have any applicable aging effect. I happen to disagree - Pilgrim Station site is located on a body of salt water. The air is therefore laden with salt crystals. Salt air can cause severe corrosion to aluminum.
2. Section 3.6.2.3, Pgs. 507, 508 - The licensee claims that fuse holders with metallic clamps are not subject to aging management review because they are part of an active device or are located in circuits that perform no intended function.
 - Other licensees have included these devices in scope. What is different at Pilgrim? How can these fuse holders be active components?

B. License Renewal Application, (LRA):

1. I noted in my review of the LRA and the audit report that the condensate storage tank and the diesel fuel oil tank sit on sand beds. The applicant has proposed a one time random ultrasonic test of these tank bottoms.
 - Industry experience with tanks on sand beds has shown that failure on tank bottoms has occurred. Is the proposed one time inspection the correct aging management program?
2. It was noted in various sections of the LRA, that the applicant has taken credit for the surveillance program and the preventing maintenance program for managing the aging of various components.
 - Has the staff reviewed details of the station preventive maintenance program to ensure that the program in fact addresses what the licensee claims? Specifically, has the staff evaluated program procedures that cover aging management program B.1.24 "Periodic Surveillance and Preventive Maintenance"?
 - Also noted that the licensee takes credit for aging management of many components via the system walkdown program. Has the staff evaluated the effectiveness of the system walkdown program as it relates to the aging management of components?

3. Electrical and Instrument and Controls Section 3.6.2.2.2 "Salt Deposits on High Voltage Insulators".
The Licensee states that salt build-up is a short term concern, therefore not an applicable aging mechanism.
 - Didn't Pilgrim Station have an event sometime in the past due to salt build-up on switchyard components?
4. Electrical and Instrument and Controls Table 3.6.2-1, the component is cable connections -
 - "Why does the licensee state that no aging management is required? - I consider cable connections as "passive".

C. Safety Evaluation Report Items:

1. What is the staff's conclusion regarding the security diesel generator in-scope items? This apparently was turned over to Region 1 to decide.
 - Has this been resolved? How?
2. Appendix A - License Renewal Commitments
 - Why are no commitments being implemented prior to June 8, 2012?

D. Miscellaneous Issues:

1. I would like to hear what the licensee and NRC staff have to say regarding status of the open items.
2. What is the staff's assessment of Pilgrim Station performance?
Are there any performance areas that are not "Green"?

A handwritten signature in black ink, appearing to read "John J. Barton". The signature is written in a cursive style with a large, looping initial "J".

From: Maitri Banerjee
To: Graham B. Wallis; happymariner@adelphia.net; jdsieber@aol.com;
mvbonaca@snet.net; omaynard@allegiance.tv; wjshack@anl.gov
Date: 03/28/2007 4:46:30 PM
Subject: Re: Pilgrim License Renewal Subcommittee Briefing on 4/4/07

Enclosed is the Pilgrim inspection report.

We have several reporters who want to listen in during the meeting.

Please let me know if you have any questions.

With regards.

Maitri

CC: Cayetano Santos

Mail Envelope Properties (460AD426.11C : 12 : 35638)

Subject: Re: Pilgrim License Renewal Subcommittee Briefing on 4/4/07
Creation Date 03/28/2007 4:46:30 PM
From: Maitri Banerjee
Created By: MXB@nrc.gov

Recipients	Action	Date & Time
adelphia.net PM happymariner (<u>happymariner@adelphia.net</u>)	Transferred	03/28/2007 4:46:56
allegiance.tv PM omaynard (<u>omaynard@allegiance.tv</u>)	Transferred	03/28/2007 4:46:56
anl.gov PM wjshack (<u>wjshack@anl.gov</u>)	Transferred	03/28/2007 4:46:56
aol.com PM JDSIEBER (<u>jdsieber@aol.com</u>)	Transferred	03/28/2007 4:46:56
Dartmouth.EDU PM Graham.B.Wallis (Graham B. Wallis)	Transferred	03/28/2007 4:46:56
nrc.gov TWGWPO02.HQGWDO01 PM CXS3 CC (Cayetano Santos)	Delivered	03/28/2007 4:46:35
AM	Opened	03/29/2007 10:36:00
snet.net PM mvbonaca (<u>mvbonaca@snet.net</u>)	Transferred	03/28/2007 4:46:56
Post Office	Delivered	Route adelphia.net allegiance.tv anl.gov aol.com

TWGWPO02.HQGWDO01

Dartmouth.EDU
03/28/2007 4:46:35 PM nrc.gov
snet.net

Files	Size	Date & Time
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ReplyRequested:	No
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To Be Delivered:	Immediate
Status Tracking:	Delivered & Opened

March 15, 2007

Mr. Kevin Bronson
Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5508

SUBJECT: PILGRIM NUCLEAR POWER STATION - NRC LICENSE RENEWAL
INSPECTION REPORT 05000293/2006007

Dear Mr. Bronson:

On October 6, 2006, an NRC team completed the onsite portion of its inspection of your application for license renewal of your Pilgrim Nuclear Power Station. One inspector returned on December 7 and 8, 2006 to inspect your resolution of an NRC-identified concern. The enclosed report documents the results of the inspection, which were discussed on January 30, 2006, with you and members of your staff in an exit meeting open for public observation in Plymouth, Massachusetts.

The purpose of this inspection was to examine the plant activities and documents that support the application for a renewed license of Pilgrim Nuclear Power Station. The inspection reviewed the screening and scoping of non-safety related systems, structures, and components, as required in 10 CFR 54.4(a)(2), and determined whether the proposed aging management programs are capable of reasonably managing the effects of aging. These NRC inspection activities constitute one of several inputs into the NRC review process for license renewal applications.

The inspection team concluded screening and scoping of non-safety related systems, structures, and components were implemented as required in 10 CFR 54.4(a)(2), and the aging management portion of the license renewal activities was conducted as described in the License Renewal Application. The inspection concluded that the documentation supporting the application was in an auditable and retrievable form. The team identified a number of areas that resulted in changes to the application, programs, and procedures.

Overall, the inspection results support a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in your application and that the intended functions of these systems, structures, and components will be maintained in the period of extended operation. However, the inspection identified one program area (drywell shell monitoring), which has resulted in an open item in the safety evaluation report (SER) issued by the Office of Nuclear Reactor Regulation (NRR) on March 1, 2007. Also, the inspection team could not reach a determination on another program (instrumentation circuits testing) due to additional program development by the industry.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at <http://www.nrc.gov/reading-rm/adams.html> (the Public Electronic Reading Room).

Sincerely,

/RA/

Richard J. Conte, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-293
License No. DPR-35

Enclosure: Inspection Report 05000293 /2006007

cc w/encl:

G. J. Taylor, Chief Executive Officer, Entergy Operations
M. Kansler, President, Entergy Nuclear Operations, Inc.
J. T. Herron, Senior Vice President
M. Balduzzi, Senior Vice President, Northeastern Regional Operations
C. Schwarz, Vice-President, Operations Support
S. J. Bethay, Director, Nuclear Safety Assurance
O. Limpas, Vice President, Engineering
J. F. McCann, Director, Licensing
C. D. Faison, Manager, Licensing
R. Patch, Director of Oversight, Entergy Nuclear Operations, Inc.
B. S. Ford, Manager, Licensing, Entergy Nuclear Operations, Inc.
T. C. McCullough, Assistant General Counsel
S. Lousteau, Treasury Department, Entergy Services, Inc.
Director, Radiation Control Program, Commonwealth of Massachusetts
C. White, RRPT, CHP, Radiological Health, Vermont Department of Health
The Honorable Therese Murray
The Honorable Vincent deMacedo
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Chairman, Nuclear Matters Committee
Plymouth Civil Defense Director
D. O'Connor, Massachusetts Secretary of Energy Resources
J. Miller, Senior Issues Manager
Office of the Commissioner, Massachusetts Department of Environmental Protection
Office of the Attorney General, Commonwealth of Massachusetts
Electric Power Division, Commonwealth of Massachusetts
R. Shadis, New England Coalition Staff

K. Bronson

3

D. Katz, Citizens Awareness Network
Chairman, Citizens Urging Responsible Energy

J. Sniezek, PWR SRC Consultant

M. Lyster, PWR SRC Consultant

W. Meinert, Nuclear Engineer

J. Muckerheide, MEMA and Commonwealth of Massachusetts, SLO Designee
Commonwealth of Massachusetts, Secretary of Public Safety

Distribution w/encl: (VIA E-MAIL)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No: 50-293

License No: DPR-35

Report No: 05000293/2006007

Licensee: Entergy Nuclear Operations, Inc. (Entergy)

Facility: Pilgrim Nuclear Power Station

Location: 600 Rocky Hill Road
Plymouth, MA 02360

Dates: September 18 - 22, 2006
October 2 - 6, 2006
December 6 - 7, 2006

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SUMMARY

IR 05000293/2006007; 09/18/2006 - 12/07/2006; Pilgrim Nuclear Power Station; Inspection of the Scoping of Non-Safety Systems and the Proposed Aging Management Procedures for the Pilgrim Nuclear Power Station Application for Renewed License.

This inspection of license renewal activities was performed by seven regional office inspectors. The inspection was performed in accordance with NRC Manual Chapter 2516 and NRC Inspection Procedure 71002. This inspection did not identify any findings as defined in NRC Manual Chapter 0612. The inspection team concluded screening and scoping of non-safety related systems, structures, and components were implemented as required in 10 CFR 54.4(a)(2), and the aging management program portions of the license renewal activities were conducted as described in the License Renewal Application. The inspection concluded the documentation supporting the application was in an auditable and retrievable form. The team identified a number of areas that resulted in changes to the application, programs, and procedures.

Overall, the inspection results support a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in your application and that the intended functions of these systems, structures, and components will be maintained in the period of extended operation. However, the inspection identified one program area (drywell shell monitoring), which has resulted in an open item in the safety evaluation report (SER) issued by the Office of Nuclear Reactor Regulation (NRR) on March 1, 2007. Also, the inspection team could not reach a determination on another program (instrumentation circuits testing) due to additional program development by the industry.

TABLE OF CONTENTS

	Page
SUMMARY	ii
TABLE OF CONTENTS	iii
4. OTHER ACTIVITIES (OA)	1
40A2 Other - License Renewal	1
a. Inspection Scope (IP 71002)	1
b.1. Scoping of Non Safety-Related Systems, Structures, and Components. .	1
b.2. Aging Management Programs	3
Containment Inservice Inspection Program	3
Inservice Inspection (ISI) Program	4
Containment Leak Rate Program	5
Diesel Fuel Monitoring Program	5
Lube Oil Analysis Program	6
Heat Exchanger Monitoring Program	7
Buried Piping and Tanks Inspection Program	8
Structural Monitoring Program	8
Masonry Wall Program	10
Water Control Structures Monitoring Program	10
Instrument Air Quality Program	11
Periodic Surveillance and Preventive Maintenance Program	11
Service Water Integrity Program	12
Metal-Enclosed Bus Inspection Program	13
Non-EQ Insulated Cables and Connections Program	13
Non-EQ Instrumentation Circuits Test Review Program	14
Non-EQ Inaccessible Medium - Voltage Cable	14
One-Time Inspection Program	15
Selective Leaching Program	15
System Walkdown Program	16
Flow Accelerated Corrosion Program	16
Water Chemistry Control Programs	17
Boiling Water Reactor Feedwater Nozzle Program	18
Boiling Water Reactor Stress Corrosion Cracking Program	19
Fire Protection Program	20
Fire Water System Program	20
b.3. System Review	21
Reactor Building Closed Cooling Water System Review	21
High Pressure Coolant Injection System Review	21
c. Overall Conclusions	22
40A6 Meetings, Including Exit	22
SUPPLEMENTAL INFORMATION	A-1
KEY POINTS OF CONTACT	A-1
LIST OF DOCUMENTS REVIEWED	A-1
LIST OF ACRONYMS	A-10

Report Details

4. OTHER ACTIVITIES (OA)

4OA2 Other - License Renewal

a. Inspection Scope (IP 71002)

This inspection was performed by NRC Region I based inspectors to evaluate the thoroughness and accuracy of the screening and scoping of non-safety related systems, structures, and components, as required in 10 CFR 54.4(a)(2) and to evaluate whether aging management programs will be capable of managing identified aging effects in an appropriate manner.

The inspectors selected a number of systems, components, and structures for review to determine if the methodology applied by the applicant appropriately addressed the non-safety systems affecting the safety functions of a system, structure, or component within the scope of license renewal.

The inspectors selected a sample of aging management programs to verify the adequacy of Entergy's guidance, implementation activities, and documentation. The selected aging management programs were reviewed to determine whether the proposed aging management implementing processes would adequately manage the effects of aging.

The inspectors reviewed supporting documentation and interviewed applicant personnel to confirm the accuracy of the license renewal application conclusions. For a sample of plant systems and structures, the inspectors performed visual examinations of accessible portions of the systems to observe aging effects.

b.1. Scoping of Non Safety-Related Systems, Structures, and Components

For scoping and screening, the inspectors reviewed Entergy's program guidance procedures and summaries of scoping and screening results for Pilgrim to assess the thoroughness and accuracy of the methods used to bring systems, structures, and components within the scope of license renewal into the application and to screen non-safety related systems, structures, and components, as required in 10 CFR 54.4(a)(2). The inspectors determined Entergy's procedures to be consistent with the NRC accepted guidance in Sections 3, 4, and 5 of Appendix F to NEI 95-10, Revision 5 (3: non-safety related systems, structures, and components within scope of the current licensing basis, 4: non-safety related systems, structures, and components directly connected to safety-related systems, structures, and components, and 5: non-safety related systems, structures, and components not directly connected to safety-related systems, structures, and components).

The inspectors reviewed the set of license renewal drawings, which had been color-coded to indicate systems and components in scope for 10 CFR 54.4.(a)(1) and (a)(3). The inspectors interviewed personnel, reviewed license renewal program documents,

Enclosure

and independently inspected numerous areas within the plant to confirm that appropriate systems, structures, and components had been included within the license renewal scope, that systems, structures, and components excluded from the license renewal scope had an acceptable basis, and that the boundary for determining license renewal scope within the systems, including seismic supports and anchors, was appropriate.

The in-plant areas and systems reviewed included the following:

- Reactor Building;
- Turbine Building;
- Intake Structure;
- Ventilation Stack;
- Station Blackout (SBO) Diesel Generator Building;
- Diesel Generator Rooms;
- High Pressure Coolant Injection (HPCI) System
- Reactor Building Closed Cooling Water (RBCCW) System
- Standby Liquid Control (SLC) System
- Standby Gas Treatment System
- Fire Protection System;
- Nitrogen Supply System;
- Compressed Air/Instrument Air System; and
- Service Water System.

For systems, structures, and components selected regarding spatial interaction (failure of nonsafety-related components adversely affecting safety-related components), the inspectors confirmed that the in-plant configuration had been accurately and acceptably categorized within the license renewal program documents. The inspectors determined the personnel involved in the process were knowledgeable and appropriately trained.

For systems, structures, and components selected regarding structural interaction (seismic design of safety-related components dependent upon nonsafety-related components), the inspectors determined that an erroneous assumption had caused incorrect structural boundaries, which had resulted in incorrect categorization within the license renewal program documents. Specifically, the inspectors determined that Entergy had misinterpreted existing piping symbols on the piping and instrumentation drawings (P&IDs) to represent seismic design boundaries. In particular, by inspecting the configuration in the plant of the standby liquid control system and the standby gas treatment portion of the compressed air system, the inspectors determined that incorrect boundaries had resulted, which had caused some non-safety related piping and components to be incorrectly omitted from the license renewal scope.

Entergy agreed to re-evaluate questionable structural interaction boundaries using correct assumptions and to revise scoping determinations and documents.

An inspector returned on December 6 and 7, 2006, to follow up on the re-evaluated structural boundaries. Entergy had reviewed the interfaces between safety-related and non-safety related systems to determine questionable structural interaction boundaries, had re-determined the structural boundaries based on system seismic analyses, had walked down plant configurations to confirm accurate structural boundaries, and was revising program documents to reflect the correct boundaries.

The inspector reviewed the results of the re-evaluation, reviewed isometric drawings used to determine boundaries, interviewed the structural engineer and system engineer involved in the re-evaluations, and visually inspected a majority of the applicable plant configurations. The inspector concluded that the re-evaluation had produced accurate scoping determinations for structural interaction and that piping, supports, and components had been acceptably categorized. In addition, this area was addressed under item 586 of Attachment D to License Renewal Application Amendment 10, issued on December 12, 2006.

As revised, the inspectors concluded that Entergy had implemented an acceptable method of scoping and screening of non-safety related systems, structures, and components and that this method resulted in accurate scoping determinations.

b.2. Aging Management Programs

Containment Inservice Inspection Program

The Containment Inservice Inspection Program (American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWE Program) is an existing program modified for the purpose of aging management and is credited with managing the aging effects in primary containment systems, i.e., drywell and torus. ASME Section XI, Subsection IWE provides for inspection of primary containment components and the containment vacuum breaker system piping and components. It covers steel containment shells and their integral attachments, containment hatches and airlocks, seals and gaskets, containment vacuum breaker system piping and components, and pressure retaining bolting. The aging effects are managed by periodic visual inspections and periodic ultrasonic testing wall thickness measurements.

The inspectors interviewed the program owner; reviewed the existing program documents, the results (data sheets) from program inspections, and applicable corrective action documents; and performed a walkdown of the torus room to assess the condition of the outside of the torus. The inspector reviewed the commitments contained in the Pilgrim response letter to Generic Letter 87-05, "Request for Additional Information - Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells."

The inspectors evaluated whether prior leakage could have adversely affected the outside of the drywell shell. Entergy noted that there is no record of an incident or event during which leakage occurred, that the drywell design incorporates multiple drains to prevent leakage from affecting the drywell shell, including an alarmed flow switch in a

Enclosure

drain, and that ultrasonic testing (UT) measurements of the shell have confirmed nominal shell thickness. Nonetheless, the inspectors noted the following:

- On September 19, 2006, and on multiple prior occasions, water was found on the torus room floor, potentially indicative of leakage affecting the drywell.
- The flow switch failed its surveillance test in December 2005 due to clogging and had not been repaired.
- Monitoring of applicable drains has been inconclusive, in that records indicated negligible results.

The inspectors noted that NRC License Renewal Interim Staff Guidance LR-ISG-2006-01 specifies that if "moisture has been detected or suspected" on the exterior of the drywell, then augmented inspections should be performed and "surfaces accessible from one side only shall be examined for wall thinning using [UT]". During the inspection Entergy had no plans to perform future UT inspections. Subsequently, after consultation with the NRC inspection team, the Office of Nuclear Reactor Regulation (NRR) issued Request for Additional Information (RAI) B.1.16.1 on November 7, 2006 on this issue. The Entergy response is contained in Attachment B to License Renewal Application Amendment 10, issued on December 12, 2006. In addition, NRC Safety Evaluation Report (SER) Open Item (OI 3.0.3.3.2) addresses this issue. On January 29, 2007, Entergy submitted plans to perform future drywell UT inspections. Ultimate resolution of this issue will therefore be documented by NRR in a future revision to the SER.

For the Containment Inservice Inspection Program other than the drywell shell, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and plant historical reviews, to determine appropriate aging effects.

Inservice Inspection (ISI) Program

The Inservice Inspection Program (ASME Section XI, Subsection IWF Program) is an existing program credited with managing the aging effects on ASME Section XI Class 1, 2, 3 and MC components and piping support members for loss of mechanical function and loss of material. Bolting is also included with these components. The aging effects are managed by periodic visual examinations for corrosion and loss of material in structural members, and loss of preload in bolting; missing, detached, or loosened members or bolts; and any degradation of protective coatings. The program has been enhanced by including additional MC components in the approved ASME Section XI, Inservice Inspection program.

The inspectors reviewed the program description, program basis documents, the currently approved ASME Section XI, Subsection IWF program, and the results of previous inspections and examinations. The documents reviewed and discussions with cognizant individuals indicated the operating experience of the Inservice Inspection Program at Pilgrim, which includes ASME Section XI, Subsection IWF aging management activities, has not shown any adverse trend. Periodic self-assessments of the program have been performed to identify the areas that need improvement to maintain the quality and integrity of the program.

For the Inservice Inspection Program, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Containment Leak Rate Program

The Containment Leak Rate Program is an existing program which monitors the leak rate from the primary containment and complies with the regulations of 10 CFR Part 50, Appendix J. The containment leak rate tests assure that leakage through the primary containment and connected systems is within allowable limits, and periodic surveillance testing of containment penetrations and isolation valves is performed so that appropriate maintenance and repairs are performed.

The inspector reviewed the existing program and implementing procedures, interviewed the program owner, reviewed the results from the past three containment integrated leak rate tests (ILRTs), and reviewed the historical results and trends from past ILRTs over the past 34 years of operation. All regulatory requirements were met.

For the Containment Leak Rate Program, the inspectors concluded that the program has an acceptable record of complying with 10 CFR Part 50, Appendix J and that Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine the appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Diesel Fuel Monitoring Program

The Diesel Fuel Monitoring Program is an existing program that will be enhanced to manage the effects of fouling and loss of material (corrosion) of diesel fuel systems. The aging effects are managed by a combination of periodic chemistry sampling and analysis, and periodic fuel oil tank inspections. The enhancements will add selected systems within the scope of license renewal that need periodic monitoring of aging effects and were not previously included in this existing periodic monitoring program.

The existing fuel oil monitoring program, associated chemistry procedures, and analysis reports of the condition of fuel oils were reviewed to determine the effectiveness of the existing program. The inspectors also reviewed tank inspection records to verify that the results were within the acceptable range. In addition, the inspectors reviewed the schedule for implementation of the enhancements.

The inspectors identified that the fire pump diesel fuel oil tank was not included in the enhanced fuel monitoring program. To address this concern in the proposed program, Entergy committed to include periodic ultrasonic thickness measurements of the bottom surface of the fire pump diesel day tank, and establish acceptance criteria and inspection locations by engineering analysis. The first inspection will occur prior to the period of extended operation. Subsequent test intervals will be determined based on the results of the first inspection. Subsequently, this commitment was addressed under item 565 in License Renewal Application Amendment 9, issued on October 6, 2006.

The inspectors identified that the enhanced fuel monitoring program for the security diesel fuel oil only specified sampling for water content. To address this concern in the proposed program, Entergy committed to include quarterly sampling of the security diesel fuel oil for particulates, water, and sediment. Acceptance criteria will be established based on engineering evaluations. Subsequently, this commitment was addressed under item 566 in License Renewal Application Amendment 9, issued on October 6, 2006.

For the Diesel Fuel Monitoring Program, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. As amended, Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Lube Oil Analysis Program

The Lube Oil Analysis Program is an existing program that will be enhanced to manage the effects of loss of material (corrosion) in lubricating oil systems. The aging effects are managed by a combination of periodic chemistry sampling and analysis, and periodic replacement of lube oil. The enhancements will add selected systems within the scope of license renewal that need periodic monitoring of aging effects and were not previously included in the existing periodic monitoring program.

The inspectors reviewed the existing lube oil analysis program, supporting chemistry procedures, and analysis reports of the condition of lube oils to determine the effectiveness of the existing program.

The inspectors identified that the existing program methodology and acceptance criteria differed from applicable industry codes and standards, and the acceptance criteria and required actions were not contained within approved station procedures. To address this concern in the proposed program, Entergy committed to proceduralize the program elements described in LRPD-02, "Aging Management Program Analysis Program," for the Lube Oil Analysis Program prior to the period of extended operation. The program elements will include appropriate acceptance criteria and corrective actions in the event that acceptance criteria are not satisfied. The periodic sampling will include the parameters specified in NUREG-1801, Section XI.M39, "Lubricating Oil Analysis." In addition, Entergy committed to provide a documented basis for the acceptance criteria. Subsequently, these commitments were addressed under items 553 and 589 of Attachment D to License Renewal Application Amendment 10, issued on December 12, 2006.

During the inspection Entergy had not documented the actions to be implemented in detail. Therefore, the inspectors could not fully assess the effectiveness of the proposed program enhancements.

Enclosure

For the Lube Oil Analysis Program the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. As amended, Entergy provided adequate guidance to ensure aging effects will be appropriately identified and addressed.

Heat Exchanger Monitoring Program

The Heat Exchanger Monitoring Program is a new program under development at Pilgrim. The program is credited with managing the aging effects of loss of material (corrosion) for heat exchanger tubes, heads, covers, and tube sheets, for those heat exchangers within the scope of license renewal that need periodic monitoring of aging effects and are not covered by other existing periodic monitoring programs. The aging effects will be managed by periodic visual inspections and non-destructive examinations of selected heat exchangers in the program.

The inspectors identified that the proposed program did not specify the criteria to determine which heat exchangers within the program's scope would be selected for inspection. In addition, the proposed program did not identify inspection frequency or acceptance criteria bases. To address this concern in the proposed program, Entergy committed to select heat exchangers for inspection based on the materials of construction and associated environments, as well as the type of heat exchanger. At least one heat exchanger of each type, material, and environment combination will be included in the selected population to be examined. In addition, Entergy committed to perform eddy current examinations at a frequency consistent with applicable industry standards and evaluate the results against acceptance criteria determined by component specific engineering evaluations. Subsequently, these commitments were addressed under items 526 and 583 of Attachment D to License Renewal Application Amendment 10, issued on December 12, 2006.

The program, as revised, will provide inspection and acceptance criteria, and will evaluate the inspection results. Inspections will be performed in accordance with approved station procedures. Inspection methods will include visual examinations on accessible heat exchanger heads, covers, and tube sheets, as well as eddy current non-destructive examinations of tube wall thickness, where practical. The examinations will be performed at a frequency determined by internal and external operating experience. The program will be initiated prior to the period of extended operation. Based on review of the proposed scope, parameters to be monitored, method of monitoring, and acceptance criteria, the inspectors determined that the proposed heat exchanger monitoring program, when implemented as planned, will provide assurance that heat exchangers are routinely evaluated for age-related degradation of loss of material.

For the Heat Exchanger Monitoring Program, the inspectors concluded that Entergy had performed adequate evaluations, including industry experience and historical reviews, to determine appropriate aging effects. As amended, Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Enclosure

Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program is a new program credited with managing the loss of material aging effects on the external surfaces of piping in a soil environment, including the service water, standby gas treatment, and condensate transfer systems. The aging effects are to be managed by preventive measures, i.e., coatings and wrappings, and inspections to manage the effects of corrosion on component surfaces. As described in Appendix B, Part 1.2 of the application, Entergy plans to perform a focused inspection within the first 10 years of the period of extended operations.

The inspectors reviewed the program basis document, system drawings, and documented corrective action reviews, and interviewed the responsible plant personnel regarding these documents and inspection criteria to be used. Also, the inspectors walked down the service water, standby gas treatment, and condensate transfer systems in the vicinity of buried piping. The inspectors noted that the new program would not have any inspections until the period of extended operation. The inspectors discussed this with Entergy, and they agreed to perform at least one inspection of buried piping prior to the period of extended operation and to clarify this within program procedures.

For the Buried Piping and Tanks Inspection Program, the inspectors concluded that Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. As revised, Entergy provided adequate guidance to ensure the aging effects are appropriately identified and addressed.

Structural Monitoring Program

The Structural Monitoring Program is an existing program that has been modified and will be further enhanced to include additional structures and structural components, including structural bolting, into the program. The program was developed based on guidance in Regulatory Guide 1.160 Revision 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and NUMARC 93-01 Revision 2, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," to satisfy the requirement of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The scope of the program also includes condition monitoring of masonry walls and water-control structures as described in the Masonry Wall Program and of Water-Control Structures Monitoring Program (discussed separately below). The enhancements to the program include the addition of items (such as seals, gaskets, seismic joint fillers, roof elastomers) that are not currently monitored but need monitoring during the period of extended operation.

Aging effects are managed by periodic visual inspections by qualified personnel to monitor structures and components for applicable aging effects. Specifically, concrete structures are inspected for loss of material, cracking, and a change in material properties. Steel components are inspected for loss of material due to corrosion.

Enclosure

Component supports will be inspected for loss of material, reduction or loss of isolation function, and reduction in anchor capacity due to local concrete degradation. Exposed surfaces of bolting are monitored for loss of material (corrosion), loose nuts, missing bolts, or other indications of loss of preload.

The inspectors reviewed the program description, program basis documents, approved station procedures, and results of prior inspections, held discussions with cognizant personnel, and performed a walkthrough visual examination of accessible structural items, including reinforced concrete and structural steel members, components and systems to assess the effectiveness of the current program. The inspectors reviewed station procedures, maintenance history, prior inspection findings and followup, and current inspection schedules. Inspection frequency is every three (3) years for accessible areas, and every ten (10) years for normally inaccessible areas. The program contains provisions for more frequent inspections to ensure that observed conditions that have the potential for impacting an intended function are evaluated or corrected in accordance with the corrective action process. The Structures Monitoring Program is consistent with the ten elements of aging management program XI.S6, "Structures Monitoring Program," specified in the Generic Aging Lessons Learned (GALL) Report (NUREG 1801).

The inspectors noted that the structural monitoring inspections documented the acceptability of structures but did not address the applicable conditions of the structure which existed. The lack of condition information would prevent trending and monitoring of any condition, which though acceptable, might further degrade. To address this concern in the proposed program, Entergy stated that this insight would be incorporated within procedural changes underway to upgrade procedural guidance for all structural monitoring programs to provide sufficient descriptions of conditions during inspections to enable effective monitoring and trending.

The inspectors noted that Entergy plans to inspect normally inaccessible areas every 10 years. Nonetheless, some areas of the concrete foundation of the plant stack (a safety-related structure) have not been inspected due to radiation concerns (ALARA considerations), and there were no plans to inspect the foundation. While the inspectors agreed that ALARA considerations could prevent periodic inspections, an inspection was appropriate. To address this concern in the proposed program, Entergy committed to perform an inspection of the previously uninspected areas of the plant stack foundation. Subsequently, this commitment was addressed under item 581 of Attachment D to License Renewal Application Amendment 10, issued on December 12, 2006.

For the Structural Monitoring Program, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. As amended, Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Enclosure

Masonry Wall Program

The Masonry Wall Program is an existing program, which is a part of the existing Structural Monitoring Program. The Masonry Wall Program is credited with managing the aging effects in masonry walls. The aging effects are managed by a program of inspection of masonry walls for cracking on a frequency of 3 years to assure that the established evaluation basis for each masonry wall remains valid during the period of extended operation. Elastomers will be monitored for change in material properties.

The inspectors reviewed the program description, the program basis documents, the approved station procedures, and the results of prior inspections; held discussions with cognizant personnel; and performed a walkthrough visual examination of accessible masonry walls to assess the effectiveness of the current program. The scope of the program includes all masonry walls that perform intended functions in accordance with 10 CFR 54.4, and were covered by I. E. Bulletin 80-11 "Masonry Wall Design." The inspections are implemented through station procedures. Maintenance history has not revealed any degradation (such as cracks) of masonry block walls, providing evidence that the walls continue to provide their intended design function. In response to I.E. Bulletin 80-11, and Information Notice 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to I.E. Bulletin 80-11," various actions have been taken, including program enhancements and follow-up inspections to substantiate masonry wall analyses and classifications. These actions have addressed all concerns raised by I.E. Bulletin 80-11 and Information Notice 87-67, namely, unanalyzed conditions, improper assumptions, improper classification, and lack of procedural controls. A review of operating experience at Pilgrim indicated that the program is effective for managing aging effects of masonry walls.

For the Masonry Wall Program, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Water Control Structures Monitoring Program

The Water Control Structures Monitoring Program is an existing program modified for the purpose of aging management program and is credited with managing the aging effects in water control structures and systems. The aging effects are managed by periodic inspections of the water control structures for structural and hydraulic degradation, and potential loss of function for intended service. The water control structure aging management program is a subpart of the main Structures Monitoring Program. It is based on the guidance provided in RG 1.127 and ACI 349.3R and will provide for periodic inspection of water control structures (breakwaters, jetties, and revetments). The program will be used to manage loss of material, cracking, and change in material properties for concrete components; loss of material and change in material properties for wooden components; and loss of material and loss of form (such as subsidence, settlement, leaks, and bowing) for dikes and embankments. Inspection

Enclosure

frequency is every five (5) years. The program will be enhanced to ensure that water-control structural aging effects are adequately managed during the period of extended operation.

The inspectors reviewed the program description, program basis documents, approved station procedures, and results of prior inspections; interviewed cognizant personnel; and visually examined accessible water control structures to assess the effectiveness of the current program. As the Water Control Structures Monitoring Program is a subpart of the larger structural monitoring program, this review was performed in conjunction with the comprehensive review of the main Structural Monitoring Program. The Water Control Structures Monitoring Program is consistent with the ten elements of aging management program XI.S7, "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants," specified in the GALL Report.

For the Water Control Structures Monitoring Program, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Instrument Air Quality Program

The Instrument Air Quality Program is an existing program that will be enhanced to manage the effects of loss of material (corrosion) on components supplied with instrument air by ensuring that the instrument air is free of water and significant contaminants. The aging effects are managed by periodic samples for dew point, oil mist, and particulate. The existing program will be enhanced to include additional sample locations in the instrument air system at susceptible locations.

The inspectors reviewed the existing Pilgrim procedures for instrument air quality sampling, the proposed enhanced sampling points, as well as recent sample results to verify that results were within the acceptable range.

For the Instrument Air Quality Program, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Periodic Surveillance and Preventive Maintenance Program

The Periodic Surveillance and Preventive Maintenance Program is an existing program that will be expanded to include selected systems within the scope of license renewal that need periodic monitoring of aging effects and are not covered by other existing periodic monitoring programs. The program will verify the integrity of the systems and manage any identified aging effects. The program will manage the aging effects of change in material properties, loss of material, and reduction of heat transfer capability for various systems and environments. The aging effects will be managed by periodic condition monitoring examinations performed at susceptible locations in the systems,

Enclosure

intended to assure that existing environmental conditions are not causing material degradation that could result in a loss of system intended functions. The periodic inspections will be performed on a pre-determined frequency, typically once every 5 years, during the period of extended operation.

This program will provide inspection and acceptance criteria, and will evaluate the inspection results. Inspections will be performed in accordance with station procedures that are based on applicable codes and standards. Inspection methods may include visual examinations of disassembled components or volumetric non-destructive examination techniques. The inspectors reviewed the proposed periodic inspection table, which contained a listing of selected systems and components to be periodically inspected, parameters to be monitored, and inspection techniques, and acceptance criteria. Based on review of the proposed program, the inspectors determined that the Periodic Surveillance and Preventive Maintenance Program, when implemented as planned, will provide assurance that systems and components are routinely inspected for age related degradation of change in material properties, loss of material, and reduction of heat transfer capability for systems, components, and environments, and will adequately manage the identified aging effects.

For the Periodic Surveillance and Preventive Maintenance Program, the inspectors concluded Entergy had performed adequate evaluations, including industry experience and historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Service Water Integrity Program

The Service Water Integrity Program is an existing program credited with managing the maintenance of protective coatings and piping as well as the prevention of excess biofouling associated with the open cycle service water system. The aging effects are managed by surveillance tests and control techniques addressed by NRC Generic Letter 89-13 to manage the effects of material loss and fouling.

The inspectors reviewed the present program and its implementing procedures, interviewed the program owner, performed a walkdown of accessible portions of the service water system, reviewed service water system health reports, and reviewed numerous past instances where the corrective action system had identified and fixed aging degradation in the system.

For the Service Water Integrity Program, the inspectors concluded that Entergy had performed adequate evaluations, as well as industry experience and plant historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects would be appropriately identified and addressed.

Metal-Enclosed Bus Inspection Program

The Metal-Enclosed Bus Inspection Program is a new program created for license renewal and is credited with managing the aging effects on the non-segregated phase bus, which connects 4.16kV switchgear. The purpose of the program is to visually inspect the enclosure assemblies and interior portions of the bus for structural integrity, signs of cracking, and water or debris.

At the time of the inspection, Entergy had just begun implementation of the actions identified in the program. Therefore, the inspector could not assess the implementation of this program. The inspector reviewed drawings, procedures and performed walkdowns to assure the proposed program will be capable of managing the aging effects. The inspector also had discussions with system engineers to determine what will be in scope of the proposed inspection program and the current conditions of the non-segregated bus.

For the Metal Enclosed Bus Inspection Program, the inspector concluded that Entergy performed adequate evaluations, which included industry experience and historical reviews, to determine appropriate aging effects. In program-level documents, Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Non-EQ Insulated Cables and Connections Program

The Non-EQ Insulated Cables and Connections Program is a new program created for license renewal and is credited with managing the aging effects of electrical cables and connections exposed to an adverse environment, but not subject to 10 CFR 50.49, "Environmental Qualification Requirements." The goal of this program is to provide reasonable assurance that the intended functions of these insulated cables and connections can be maintained through the period of extended operation.

At the time of the inspection, Entergy had not completed any of the actions identified in the program. Therefore, the inspector could not assess the effectiveness of the implementation of this program. The inspector reviewed drawings, performed walkdowns, and had discussions with system engineers to assess the proposed program and assure it will be capable of managing aging effects.

For the Non-EQ Insulated Cables and Connections Program, the inspector concluded that Entergy performed adequate evaluations, which included industry experience and historical reviews, to determine appropriate aging effects. In program-level documents, Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Non-EQ Instrumentation Circuits Test Review Program

The Non-EQ Instrumentation Circuits Test Review Program is a new program created for license renewal and is credited with managing the aging effects of electrical cables and connections exposed to an adverse environment, but not subject to 10 CFR 50.49, "Environmental Qualification Requirements." The purpose of this program is to develop a method of reviewing calibration and surveillance test results to detect aging effects in circuits with sensitive, high voltage, low-level signals. Entergy plans to perform the review by using the results from current calibrations and functional tests to determine the aging effects on this equipment.

Because the GALL Report has not established programmatic standards for this program, and the testing parameters are being developed as part of an industry-wide initiative, the inspectors were unable to review implementation documents. Nonetheless, the inspectors interviewed plant personnel to understand the current status of the program and the level of understanding of cognizant plant engineers. The inspectors reviewed current procedures and a sample of completed surveillance tests to evaluate the current status of the program.

The inspectors concluded Entergy had performed adequate evaluations, which included industry experience and plant historical reviews, to determine appropriate aging effects. Because there is incomplete guidance in the GALL Report, the essential test parameters have not been established, and the aging management program is still being developed, the inspectors were unable to determine the acceptability of the proposed aging management program.

The team discussed this matter with the NRC program staff in the NRC Office of Nuclear Reactor Regulation; resolution of this issue will be coordinated between the NRC Region I and NRR offices.

Non-EQ Inaccessible Medium - Voltage Cable

The Non-EQ Inaccessible Medium-Voltage Cable Program is a new program created for license renewal which is credited with managing the aging effects of inaccessible 2KV to 35KV cables not subject to 10 CFR 50.49, "Environmental Qualification Requirements," and exposed to an adverse environment. The purpose of this program is to periodically inspect inaccessible cables for exposure to significant moisture and to evaluate the condition of the conductor insulation if exposed to significant moisture.

At the time of the inspection, Entergy had not completed any of the actions identified in the program. While the inspector was not able to thoroughly assess the effectiveness of the implementation of the program as a whole, Entergy does have a preliminary procedure in place for inspecting cables located in manholes. The inspector reviewed the procedure, performed walkdowns, and had discussions with system engineers on the effectiveness of the program regarding identifying age-related deficiencies.

Enclosure

For the Non-EQ Inaccessible Medium - Voltage Cable Program, the inspector concluded that Entergy performed adequate evaluations, which included industry experience and historical reviews, to determine appropriate aging effects. In program-level documents, Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

One-Time Inspection Program

The One-Time Inspection Program is a new aging management program intended to verify the effectiveness of other aging management programs, including Water Chemistry Control Programs, Nonsafety-Related Systems and Components Affecting Safety-Related Systems Program, and the Selective Leaching Program, by reviewing various aging effects for impact. Where corrosion resistant materials and/or non-corrosive environments exist, the One-Time Inspection Program is intended to verify that an aging management program is not needed during the extended period of operation by confirming that aging effects do not affect the safety functions of systems, structures, and components. Non-destructive evaluation will be performed by qualified personnel using procedures and processes consistent with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME) and 10 CFR 50, Appendix B. The One-Time Inspection Program will be implemented prior to the period of extended operation.

The inspectors reviewed the program description, implementation plan, and inspection sample basis, and discussed the planned activities with the responsible staff.

For the One-Time Inspection Program, the inspectors concluded Entergy had performed adequate evaluations, including reviews of industry experience and plant history, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Selective Leaching Program

The Selective Leaching Program is a new program credited with ensuring the integrity of components made of cast iron, bronze, brass, and other alloys exposed to raw water, brackish water, treated water, or groundwater that may lead to selective leaching. The program will include a one-time visual inspection and hardness measurement of selected components that may be susceptible to selective leaching to determine whether loss of material due to selective leaching is occurring, and whether the process fluids will affect the ability of the components to perform their intended functions for the period of extended operation. The Selective Leaching Program will be implemented prior to the period of extended operation.

The inspectors reviewed the program description, reviewed the implementation plan, and discussed the planned activities with the responsible staff.

For the Selective Leaching Program, the inspectors concluded Entergy performed adequate evaluations, including reviews of industry experience and plant history, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

System Walkdown Program

The System Walkdown Program is an existing program that will be enhanced for the purpose of aging management. The program entails inspections of external surfaces of components subject to aging management. The program is also credited with managing loss of material from internal surfaces for situations in which internal and external material and environment combinations are the same, such that external surface condition is representative of internal surface condition. System Walkdown Program guidance documents will be enhanced to include inspections of areas surrounding the subject systems to identify hazards to those systems.

The inspectors accompanied a system engineering manager during a walkdown of several systems. In addition, the inspectors reviewed station procedures, reviewed condition reports generated as a result of system walkdowns, and performed interviews to assess the effectiveness of system walkdowns and to verify that system engineers' awareness of new system walkdown guidance due to procedure enhancements being made for license renewal.

For the System Walkdown Program, the inspectors concluded Entergy had performed adequate evaluations, including reviews of industry experience and plant history, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Flow Accelerated Corrosion Program

The Flow-Accelerated Corrosion Program is an existing program credited with managing the corrosion aging effects in all carbon steel components in systems containing high-energy fluids carrying two-phase or single-phase fluid $\geq 2\%$ of plant operating time. The aging effects are managed by performing non-destructive examinations (e.g. ultrasonic testing) to detect wall thinning and by predicting wear rates to support the proactive replacement of system piping. In addition, the program provides for the performance of follow-up inspections to confirm predictions and to determine the need for repairs or replacements as necessary.

The inspectors reviewed the piping ultrasonic testing wall thickness results from previous inspections, condition reports related to the Flow Accelerated Corrosion Program, and the 2005 Flow Accelerated Corrosion Program Focused Self-Assessment Report. In addition, the inspectors interviewed the Flow Accelerated Corrosion Program Manager and noted that recent piping replacements, initiated as a result of this program,

Enclosure

were implemented preventively due to identified flow-accelerated corrosion. In each case, the replacement piping material was more resistant to corrosion than the original piping material.

For the Flow-Accelerated Corrosion Program, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and plant historical reviews, to determine the appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Water Chemistry Control Programs

The Water Chemistry Control Programs consist of the Boiling Water Reactor (BWR), the Auxiliary Systems, and the Closed Cooling Water Programs. The aging effects are managed by monitoring and control of water chemistry to minimize contaminant concentration and mitigate loss of material. The One-Time Inspection Program describes inspections planned to verify the effectiveness of water chemistry control programs to ensure that significant degradation is not occurring, and components' intended functions are maintained during the period of extended of operation.

The Water Chemistry Control - BWR Program is an existing program with the objective of managing aging effects caused by corrosion and cracking mechanisms. The BWR Program optimizes the primary water chemistry to minimize the potential for loss of material and cracking. This is accomplished by limiting the levels of contaminant in the reactor coolant system (RCS) that could cause loss of material and cracking.

The Water Chemistry Control - Closed Cooling Water Program is an existing program which includes preventive measures that manage the loss of material, cracking, and fouling for components in closed cooling water systems, such as the reactor building closed cooling water (RBCCW), turbine building closed cooling water (TBCCW), and plant heating systems.

The Water Chemistry Control - Auxiliary Systems Program is an existing program which manages the loss of material for components exposed to treated water. The Auxiliary Systems Program activities include sampling and analysis of the stator cooling water system to minimize component exposure to aggressive environments.

The Water Chemistry Control Programs are administered in accordance with the Boiling Water Reactor Vessel and Internals Project guidelines BWRVIP-130 and Electrical Power Research Institute guideline EPRI report 1007820. The inspectors reviewed chemistry procedures and interviewed Pilgrim water chemistry specialists.

For the Water Chemistry Control Programs, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and plant historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Enclosure

Boiling Water Reactor Feedwater Nozzle Program

The Boiling Water Reactor (BWR) Feedwater Nozzle Program is an existing program that is comparable to the program described in NUREG-1801, Section XI.M5, BWR Feedwater Nozzle. This program continues enhanced inservice inspection of the feedwater nozzles in accordance with the requirements of Section XI of the ASME Code and the recommendations of the BWR Owners Group Licensing Topical Report General Electric NE-523-A71-0594 for monitoring the effects of cracking of the feedwater nozzles. To minimize thermal cycling and fatigue-induced cracking of the feedwater nozzles, Entergy performed system modifications, such as removal of nozzle stainless steel cladding, removal of feedwater blend radii flaws, and installed a triple-sleeve-double piston sparger. These preventive actions are not consistent with NUREG-1801, because a low-flow controller was not installed and the reactor water cleanup system was not rerouted. However, the NRC has accepted the conclusion that these modifications were unnecessary as documented in a safety evaluation report. The aging effects of cracking are managed by specified periodic ultrasonic testing (UT) inspections of the feedwater nozzles.

To verify the effectiveness of BWR Feedwater Nozzle Program, the inspectors interviewed plant personnel and reviewed various documents, such as Plant Design Change Request PDCR 79-41, which installed a triple-sleeve-double piston sparger, selected ISI station procedures, UT feedwater nozzle examination data sheets for feedwater nozzle inside-radius and inside-radius extension and feedwater nozzle-to-shell weld for feedwater nozzles N4A through N4D from refueling outage 14 (May 2003), various ISI program health reports, condition reports, and ISI audit in Quality Assurance audit report QA-08-2005-PNP-01. Entergy changed to risk-informed ISI methodology for Class 1 piping weld inspections in the second period of the third ten-year ISI interval and performs UT inspections at intervals in accordance with Table 6.1 of GE NE-523-A71-0594. The ISI program's third ten-year interval ended on June 30, 2005 and the ISI Program Plan is in its fourth ten-year interval. Continued successful UT examination results, the most recent in 2003, have resulted in no recordable indications in the feedwater nozzles. Entergy has committed to perform an updated feedwater nozzle fatigue analysis prior to the period of extended operation as documented in License Renewal Amendment 5, dated July 19, 2006.

For the BWR Feedwater Nozzle Program, the inspectors concluded Entergy has implemented sufficient modifications and performed periodic UT inspections of the feedwater nozzles which resulted in no recordable indications, and as a consequence, the absence of indications on the feedwater nozzles provides evidence that the potential cracking of the feedwater nozzles has been reasonably managed. In addition, Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Enclosure

Boiling Water Reactor Stress Corrosion Cracking Program

The Boiling Water Reactor (BWR) Stress Corrosion Cracking Program is an existing aging management program comparable to the program described in NUREG-1801, Section XI.M7, BWR Stress Corrosion Cracking that will be enhanced prior to the period of extended operation. Entergy will enhance the implementing procedure for ASME Section XI Code Programs to specify that the guidelines in Generic Letter 88-01 or approved Boiling Water Reactor Vessel Internal Inspection Program (BWRVIP-75) shall be considered in determining sample expansion if indications are found in welds addressed by GL 88-01. The BWR Stress Corrosion Program is credited with managing crack initiation and growth due to intergranular stress corrosion cracking (IGSCC) in stainless steel piping four inches and greater nominal diameter containing reactor coolant at a operating temperature greater than 200°F or cast austenitic stainless steel (CASS). The program also applies to pump casings, valve bodies and reactor vessel attachments and appurtenances, such as vent components. The aging effects of cracking are managed by preventive measures to mitigate IGSCC, which include monitoring and controlling water impurities by water chemistry control program activities, by providing replacement piping and components that are resistant to IGSCC, and by ultrasonic examination to detect IGSCC flaws and flaw evaluation to monitor IGSCC and its effects on the reactor coolant pressure boundary components made of stainless steel or CASS. UT examinations will be performed in accordance with PNPS-RPT-05-001, Fourth Ten-Year Inspection Interval In-service Inspection (ISI) Program Plan which incorporates the guidance in NUREG-0313, NRC Generic Letter 88-01, and BWRVIP-75.

Entergy replaced the reactor recirculation and connecting portions of core spray, reactor water cleanup, and residual heat removal system piping with intergranular stress corrosion cracking resistant piping material in the 1980s. However, some non-resistant welds remain in Class 1 piping and are inspected in accordance with PNPS-RPT-05-001, GL 88-01 and BWRVIP-75 requirements. To further mitigate the initiation and propagation of IGSCC, Entergy implemented hydrogen water chemistry in the 1990s and noble metals chemical addition is planned for 2007. The inspectors reviewed the in-service inspection summary report for refueling outage (RFO) 14, ISI program health reports, ISI audit report QA-08-2005-PNP-01, various non-destructive examination data sheets, and condition reports. Based on this review the inspectors determined that no recordable indications of intergranular stress corrosion cracking have been detected during UT examinations of nozzle safe end welds and austenitic stainless steel reactor coolant piping 4" and greater and operating temperatures greater than 200°F during RFO 14 (April, 2003) and RFO 15 (April, 2005). The inspectors determined that the BWR Stress Corrosion Program has been effective in monitoring and mitigating intergranular stress corrosion cracking in the reactor coolant boundary piping systems as a result of the preventive measures, inspections, and flaw evaluations to monitor for IGSCC.

For the Boiling Water Reactor Stress Corrosion Cracking Program, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and plant historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

Fire Protection Program

The Fire Protection Program is an existing program with some enhancements and is credited with managing the aging effects of penetration seals, fire barriers, and fire-rated doors that perform a fire barrier function. It also manages the aging effects of the diesel-driven fire pump and its fuel supply line, and the halon fire suppression system.

The inspectors reviewed the system health reports, applicable procedures, and a sample of completed surveillances to assure the current program is capable of managing aging effects. The inspector also performed a walkdown of portions of the fire protection system and had discussions with system engineers on the effectiveness of the current program regarding identifying age-related deficiencies.

For the Fire Protection Program, the inspectors concluded that Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure the aging effects are appropriately identified and addressed.

Fire Water System Program

The Fire Water System Program is an existing program with some enhancements credited with managing the loss of material due to corrosion, microbiological-induced corrosion (MIC), or biofouling of carbon steel and cast-iron components in fire protection systems exposed to water. The aging effects are managed by testing in accordance with applicable National Fire Protection Association (NFPA) codes and standards to assure functionality of systems.

The inspectors reviewed system health reports, procedures, and a sample of completed surveillance tests. The inspector also performed a walkdown of portions of the fire water system and had discussions with system engineers on the effectiveness of the current program regarding identifying age-related deficiencies.

For the Fire Water System Program, the inspector concluded that Entergy had performed adequate evaluations, as well as industry experience and historical reviews, to determine appropriate aging effects. Entergy provided adequate guidance to ensure the aging effects are appropriately identified and addressed.

b.3. System Review**Reactor Building Closed Cooling Water System Review**

The Pilgrim license renewal application listed a number of plant systems within the scope of license renewal. From this list the inspectors selected the reactor building closed cooling water (RBCCW) system for a focused review to determine whether Pilgrim's aging management programs were adequate to effectively manage aging effects related to this system. The aging effects requiring management for the RBCCW system are cracking, cracking-fatigue, fouling, loss of material, and loss of material due to wear. The following existing aging management programs are credited for managing aging effects of the RBCCW system: heat exchanger monitoring; oil analysis; periodic surveillance and preventive maintenance; system walkdown; BWR water chemistry control; closed cooling water chemistry control; selective leaching program; and service water integrity.

The inspectors interviewed the RBCCW system engineer, performed walkdowns, and reviewed various documents to verify that the programs to manage aging effects, associated with the RBCCW system, have been comprehensive and effective. Specifically, the inspectors evaluated a sample of the aging management programs by reviewing RBCCW system health reports, maintenance history review, and various condition reports.

For the various aging management programs associated with the RBCCW system, the inspectors concluded that Entergy had performed adequate evaluations, including industry experience and historical reviews, of the RBCCW system to determine the appropriate aging effects and that aging management programs properly covered the various aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

High Pressure Coolant Injection System Review

The Pilgrim license renewal application listed a number of plant systems within the scope of license renewal. From this list the inspectors selected the high pressure coolant injection system (HPCI) for a focused review to determine whether the aging management programs were adequate to effectively manage aging effects related to this system. The aging effects requiring management for the HPCI system are cracking, cracking-fatigue, fouling, loss of material, and loss of material / wear. The following existing aging management programs are credited for managing HPCI aging effects: flow-accelerated corrosion; heat exchanger monitoring; oil analysis; periodic surveillance and preventive maintenance; system walkdown; and water chemistry control-BWR program.

The inspectors interviewed the HPCI system engineer, performed walkdowns, and reviewed various documents to verify the aging effects associated with the HPCI system requiring aging management have been comprehensive and effective. Specifically, the inspectors evaluated a sample of the aging management programs by reviewing HPCI system health reports, maintenance history review, in-service inspection results from the third ten-year inspection, audit report QA-08-2005-PNP-01, and various condition reports. Entergy is currently addressing leakage past HPCI steam line drain isolation valve AO-2301-29 caused by severe internal body and plug erosion, which is documented by condition report CR-PNP-2006-00773 and repair is scheduled for RFO 16, Spring 2007.

For the various aging management programs associated with the HPCI system, the inspectors concluded Entergy had performed adequate evaluations, as well as industry experience and historical reviews of the HPCI system to determine the appropriate aging effects and that aging management programs properly covered the various aging effects. Entergy provided adequate guidance to ensure aging effects are appropriately identified and addressed.

c. Overall Conclusions

Overall, the inspection results support a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in the application and that the intended functions of these systems, structures, and components will be maintained in the period of extended operation. However, the inspection identified one program area (drywell shell monitoring), which has resulted in an open item in the safety evaluation report (SER) issued by the Office of Nuclear Reactor Regulation (NRR) on March 1, 2007. Also, the inspection team could not reach a determination on another program (instrumentation circuits testing) due to ongoing testing development. The inspection concluded that the documentation supporting the application was in an auditable and retrievable form.

40A6 Meetings, Including Exit

The inspectors presented the inspection results to Mr. Kevin Bronson, general manager, Pilgrim, and other members of his staff in an exit meeting that was open for public observation on January 30, 2007. Slides from the exit meeting are located in ADAMS file no. ML070440323. Subsequent to the exit meeting by telephone, the team leader provided the revised determination on Non-EQ Instrumentation Circuits Test Review Program to Mr. Bryan Ford on March 14, 2007. Entergy had no objections to the NRC observations. No proprietary information was provided to the inspectors during this inspection.

Enclosure

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Bechen	Senior mechanical engineer
S. Bethay	Nuclear safety assessment director
S. Burke	Senior staff engineer
L. Chan	License renewal
A. Cox	License renewal specialist
D. Ellis	Senior engineer
B. Ford	Licensing manager
T. Ivy	License renewal specialist
D. Lach	License renewal project manager
M. Landry	Fire protection system engineer
L. Loomis	Water chemistry specialist
R. Mattos	Engineering consultant
F. Mogolesko	Manager, engineering projects
F. Mulcahy	System engineer
R. Pace	Design supervisor
R. Pardee	ISI Engineer
B. Rancourt	Engineer
D. Selig	EP&C supervisor
P. Smalley	Water chemistry specialist
R. Smith	Acting site VP
B. Sullivan	Systems engineering manager
T. White	Design engineering manager
S. Woods	P&C manager

LIST OF DOCUMENTS REVIEWED

LRA Drawings

Complete Set of License Renewal Drawings:

LRA-200, P&ID Legend, Sheets 1-2, Rev. 0
LRA-203, Main Steam, Sheets 1& 3, Rev. 0
LRA-209, Condensate & Demineralized Water System, Rev. 0
LRA-210, Air Injection & Off-Gas System, Rev. 0
LRA-212, Service Water System, Sheet 1, Rev. 0
LRA-215, Reactor Bldg. Closed Cooling Water System, Sheets 1-4, Rev. 0
LRA-218, Fire Protection System, Sheets 1-9, Rev. 0

LRA-219, Diesel Generator Air Start System, Rev. 0
 LRA-220, Compressed Air System, Sheets 2 & 3, Rev. 0
 LRA-223, Diesel Oil Storage & Transfer System, Rev. 0
 LRA-227, Containment Atmospheric Control System, Sheets 1-2, Rev. 0
 LRA-231, Fuel Pool Cooling System, Rev. 0
 LRA-232, Radwaste Collection, System, Rev. 0
 LRA-239, H2 & O2 Analyzer & Leak Detection System, Sheets1-2, Rev. 0
 LRA-241, Residual Heat Removal System, Sheets1-2, Rev. 0
 LRA-242, Core Spray System, Rev. 0
 LRA-243, HPCI System, Rev. 0
 LRA-244, HPCI Turbine & Control Oil Subsystem, Sheets 1-2, Rev. 0
 LRA-245, RCIC System, Rev. 0
 LRA-246, RCIC Turbine Oil Flow, Sheets 1-2, Rev. 0
 LRA-247, Reactor Water Clean-up System, Rev. 0
 LRA-249, Standby Liquid Control System, Rev. 0
 LRA-250, Control Rod Drive Hydraulic System, Sheets 1-2, Rev. 0
 LRA-251, Recirculation Pump A/B Instrumentation, Sheets 1-2, Rev. 0
 LRA-253, Nuclear Boiler, Sheets 1-2, Rev. 0
 LRA-259, Diesel Generator Turbo Air Assist System, Rev. 0
 LRA-264, Station Blackout Diesel Generator System, Rev. 0
 LRA-265, SEP Diesel Pump Fuel Oil Transfer System, Rev. 0
 LRA-271, Emergency Diesel Generator Lube Oil System, Rev. 0
 LRA-272, Emergency Diesel Generator Jacket Water Cooling System, Rev. 0
 LRA-283, Secondary Containment Isolation Control Diagram, Rev. 0
 LRA-286, Heating, Ventilation, and Air Conditioning, Rev. 0

 LRA-288, Turbine Building Air Flow Diagram, Rev. 0
 LRA-289, Reactor Building Air Flow Diagram, Rev. 0
 LRA-292, Air Flow Diagrams - Other Buildings, Rev. 0
 LRA-294, HVAC - Standby Gas Treatment System Diagram, Rev. 0
 LRA-67-96, Diesel Generators Ventilation Systems, Rev. 0

P&ID Drawings

M-226, Miscellaneous Systems, Sheet 2, Revision 26
 M-215, Reactor Building Cooling Water System, Sheet 1, Revision 50
 M-215, Reactor Building Cooling Water System, Sheet 2, Revision 47
 M-215, Reactor Building Cooling Water System, Sheet 3, Revision 39
 M-215, Reactor Building Cooling Water System, Sheet 4, Revision 45
 M-220, Compressed Air, Sheet 1, Revision 73
 M-220, Compressed Air, Sheet 2, Revision 31
 M-220, Essential Air, Sheet 3, Revision 69
 M-220, Essential Air, Sheet 5, Revision 70
 M-223, Diesel Oil Storage and Transfer System, Revision 29
 M-241, Residual Heat Removal System, Sheet 1, Revision 81
 M-241, Residual Heat Removal System, Sheet 2, Revision 46
 M-242, Core Spray, Revision 50

M-243, HPCI, Revision 51
M-244, HPCI, Sheet 1, Revision 30
M-244, HPCI, Sheet 2, Revision 9
M-246, RCIC, Sheet 1, Revision 32
M-246, RCIC, Sheet 2, Revision 1
M-245, RCIC, Revision 35
M-265, Diesel Pump Fuel Oil Transfer System, Revision 3
M-271, Diesel Generator Lube Oil System, Revision 6
M-227, Containment Atmosphere Control System, Sheet 2, Revision 48
M-264, Station Blackout Diesel Generator Set, Revision E15
M-294, HVAC - Standby Gas Treatment System Control Diagram, Revision E26
P&ID Condensate & Demineralized Water Storage & Transfer Systems, M209, Revision E65,
12/03
P&ID Air Ejection And Off-Gas System, M210, Revision E65, 9/03
P&ID Fire Protection System, M218 Sheet 1, Revision E48, 9/03
P&ID Fire Protection System, M218 Sheet 2, Revision E45, 7/04
P&ID Fire Protection System, M218 Sheet 3, Revision E45, 1/04
P&ID Service Water System, M212 Sheet 1, Revision E87, 5/05
P&ID Diesel Oil Storage & Transfer System, M223, Revision E29, 9/03
Air Ejection & Off-Gas System Underground Vent Duct, M100-168-3, Revision
M218 Sht. 1, Fire Protection System, Rev. E48
M218 Sht. 2, Fire Protection System, Rev. E45
M218 Sht. 3, Fire Protection System, Rev. E45
M218 Sht. 4, Fire Protection Halon Sybsystems, Rev. E7
M218 Sht. 5, Fire Protection System Deluge System, Rev. E4
M218 Sht. 6, Fire Protection Deluge System, Rev. E1
M218 Sht. 7, Fire Protection System, Rev. E1
M218 Sht. 8, Fire Protection System, Rev. E6
M218 Sht. 9, Fire Protection System, Rev. E2
M218 Sht. 1, Fire Protection System, Rev. E48
M218 Sht. 2, Fire Protection System, Rev. E45
M218 Sht. 3, Fire Protection System, Rev. E45
M218 Sht. 4, Fire Protection Halon Sybsystems, Rev. E7
M218 Sht. 5, Fire Protection System Deluge System, Rev. E4
M218 Sht. 6, Fire Protection Deluge System, Rev. E1
M218 Sht. 7, Fire Protection System, Rev. E1
M218 Sht. 8, Fire Protection System, Rev. E6
M218 Sht. 9, Fire Protection System, Rev. E2
M212 Service Water System, Sheet 1, Revision E87,
M212 Service Water System, Sheet 1, Revision 88
M212 Hypochlorination System, Sheet 3, Revision E66
M212 Screen Wash System, Sheet 2, Revision 72

Isometric Drawings

M100B 1003, Nitrogen Purge Supply to Drywell and Torus, Rev. 1
M1004, sht 184, Nitrogen Makeup to Drywell and Torus, Rev. E3
M1004, sht 267, Nitrogen Purge Supply Piping, Rev. E1
M1004, sht 268, Nitrogen Purge Supply Piping, Rev. E1
M1004, sht 268, Nitrogen Purge Supply Piping, Rev. E1
M1004, sht 488, Standby Gas Treatment Damper Air Supply, Rev. 1
M1004, sht 513, Vacuum Breaker Accumulator Charging, Rev. A
6498-688, Standby Liquid Control Pump Suction, Rev. E0
SKM-99-12-JCR-06, sht 1, Backup Air Supply for A EDG Damper Control, Rev. A
SKM-99-12-JCR-07, sht 1, Backup Air Supply for B EDG Damper Control, Rev. A

Other Drawings

Dwg. H-11-1-18, Revision E0, 3/6/87; Pipe Support Stand-By Liquid Control System
Dwg. H-11-1-30, Revision E2, 5/00; Pipe Support Stand-By Liquid Control System
Dwg. ISI 2R-A, Revision E3; Nuclear Boiler System, Reactor Recirc Piping Loop "A"; ISI Weld Map
Dwg. ISI 2R-A, Revision E5; Nuclear Boiler System, Reactor Recirc Piping Loop "A"; ISI Weld Map
Dwg. C21, Sh. 2, Revision E1, 4/00; Yard Work Underground Piping Sections (SSW Piping)
Dwg. C21, Sh. 3, Revision E1, 4/00; Yard Work Underground Piping Sections (SSW Piping)
Drawing MIB-45-1, Rev. E1, Reactor Vessel Feedwater Sparger
E1 Sht. 1, Single Line Diagram Station, Rev. E20
E7, Single Line Meter and Relay Diagram 4160 Volt System, Rev. E25
E1 Sht. 1, Single Line Diagram Station, Rev. E20
E6 Sht. 1, Single Line Meter & Relay Diagram Generator & Auxiliary Transformers, Rev. E16
E6 Sht. 2, Single Line Meter & Relay Diagram Generator & Auxiliary Transformers, Rev. E0
E7 Sht. 0, Single Line Meter and Relay Diagram 4160 Volt System, Rev. E25
29011 Sht. 2, Structural Details Cable Terminal House & Manhole, Rev. E1
E1 Sht. 1, Single Line Diagram Station, Rev. E20
E6 Sht. 1, Single Line Meter & Relay Diagram Generator & Auxiliary Transformers, Rev. E16
E6 Sht. 2, Single Line Meter & Relay Diagram Generator & Auxiliary Transformers, Rev. E0
E7 Sht. 0, Single Line Meter and Relay Diagram 4160 Volt System, Rev. E25
E344, Sht. 1, Electrical Plot Plan Site, Rev. 17
E344, Sht. 6, Appendix "R" Ductline General Arrangement & Parts Plan, Rev. E1
C332A, Misc. Structures - Stack Plan, Elevation and Details, Rev. E9

License Renewal Program and Procedures

LRPD-01, System and Structure Scoping Report, Rev. 0
LRPD-02, LR Aging Management Program Evaluation Report, Rev. 3
LRPD-03, LR Time-Limited Aging Analysis and Exemption Evaluations, Rev. 0
LRPD-05, Operating Experience Review Report, Rev. 0
LRPD-06, LR Time-Limited Aging Analysis - Mechanical Fatigue, Rev. 0
LRPG-03, System and Structure Scoping Methodology, Rev. 2
LRPG-05, Electrical System Scoping, Screening, and Aging Management Reviews, Rev. 2

Program Basis Documents

AMPER 3.1: Buried Piping and Tanks Program
B.1.18/AMPER 3.3: Metal-Enclosed Bus Inspection Program
B.1.20/AMPER 3.5: Non-EQ Instrumentation Circuits Test Review Program
B.1.21/AMPER 3.6 Non-EQ Insulated Cables and Connections Program
B.1.23/AMPER 3.7, One-Time Inspection Program
B.1.27/AMPER 3.8, Selective Leaching Program
B.1.1.14/AMPER 4.13, Flow Accelerated Corrosion Program
B.1.30/AMPER 4.22, System Walkdown Program
B.1.32.1/AMPER 4.23.1, Water Chemistry Control Auxiliary Systems Program
B.1.32.2/AMPER 4/23.2, Water Chemistry Control BWR Program
B.1.32.3/AMPER 4.23.3, Water Chemistry Control Closed Cooling Water Program
B.1.13.1/AMPER 4.13.1: Fire Protection Program
B.1.13.2/AMPER 4.13.2: Fire Water System Program
AMPER 4.14.2 Containment Inservice Inspection (CII) Program
AMPER 4.21 Service Water Integrity Program

Aging Management Review Technical Basis Documents

AMRE-01, Electrical Screening and Aging Management Reviews, Rev. 2
AMRM-02, Aging Management Review of Residual Heat Removal System, Revision 0
AMRM-03, Aging Management Review of Core Spray System, Rev. 0
AMRM-05, Aging Management Review of HPCI System, Rev. 0
AMRM-06, Aging Management Review of RCIC System, Rev. 0
AMRM-12, Aging Management Review of RBCCW System, Rev. 1
AMRM-13, Aging Management Review of Emergency Diesel Generator System, Rev. 0
AMRM-14, Aging Management Review of Station Blackout Diesel Generator System, Rev. 0
AMRM-15, Aging Management Review of Fuel Oil System, Rev. 0
AMRM-16, Aging Management Review of Instrument Air System, Rev. 0
AMRM-17, Aging Management Review of Fire Protection Water System, Rev. 0
AMRM-18, Aging Management Review of Halon System, Rev. 0
AMRM-30, Aging Management Review of Nonsafety-related Systems and Components
Affecting Safety-related Systems, Rev. 1
AMRM-30, Aging Management Review of Nonsafety-related Systems and Components
Affecting Safety-related Systems, Rev. 2

Plant Procedures

7.1.107, Revision 8, "Lubricating Oil Analysis using Oilview Model 5100 System"
3.M.4-123, Revision 1, "Diesel Fire Pump Engine Maintenance"
3.M.3-23.1, Revision 6, "Security Diesel System Maintenance"
3.M.4-17.4, Revision 26, "Lubrication Sampling and Change"
7.8.1, Revision 42, "Chemistry Sample and Analysis Program"
NOP02E1, Revision 2, "Generic Letter 89-13 Service Water Maintenance & Testing"
ENN-DC-120, Rev. 1, ASME Section XI Code Programs

EN-EP-S-013-P, Rev. 0, Control of Pilgrim Augmented Examinations
 ENN-CS-S-008, Pipe Wall Thinning Structural Evaluation, Rev. 0
 ENN-DC-143, System Health Reports, Rev. 2
 ENN-DC-178, System Walkdowns, Rev. 1
 ENN-DC-315, Flow Accelerated Corrosion Program, Rev. 0
 EN-LI-102, Corrective Action Process, Rev. 4
 EN-MS-S-011-Multi, Conduct of System Engineering, Rev. 0
 ENN-NDE-9.05, Ultrasonic Thickness Examination, Rev. 0
 Procedure No. 2.2.103, State Cooling System, Rev. 30
 Procedure No. 7.8.1, Chemistry Sample and Analysis Program, Rev. 40
 Procedure No. 7.8.7, Recording and Trending of Chemistry Data, Rev. 1
 8.2.12, Revision 15, "Instrument Air Header Moisture Check"
 7.1.69, Revision 14, "Air Quality Sampling"
 3.M.4-123, Diesel Fire Pump (P-140) Engine Maintenance, Rev. 1
 8.B.17.1, Inspection of Fire Door Assemblies, Rev. 17
 8.B.22, Halon 1301 System – Cable Spreading Room, Rev. 2
 Completed 1/17/04
 8.B.29, Inspection of Fire Barriers, Rev. 7
 8.B.3.1, Fire Hose Station Equipment Inspection – FSAR Related, Rev. 14
 8.B.8, Fire Hydrant Operability, Rev. 16
 Completed 6/3/04
 8.B.12, Fire Protection System Flow Tests, Rev. 29
 Completed 11/2/04
 8.B.15, Annual Functional Tests of Fire Pumps – P-135, P-140, and P-181, Rev. 28
 Completed 7/21/04
 8.7.1.4.2, Primary Containment Integrated Leak Rate Test, Revision 8, completed on 8/4/91
 8.7.1.4.2, Primary Containment Integrated Leak Rate Test, Revision 10, 5/17/93, completed on
 5/20/93
 8.7.1.4.2, Primary Containment Integrated Leak Rate Test, Revision 12, completed on 6/2/95
 8.7.1.3, Local Leak Rate Test Program, Revision 21, 10/05
 8.7.1.3.1, Performance-Based Leakage Testing Of The Primary Containment, Revision 2, 10/05
 8.5.3.14.1, Revision 2, 3/24/95; RBCCW Heat Exchanger Thermal Performance Test,
 Completed in 1995
 8.5.3.14.1, Revision 3, 2/11/97; RBCCW Heat Exchanger Thermal Performance Test,
 Completed in 1997
 8.5.3.14.1, Revision 3, 2/11/97; RBCCW Heat Exchanger Thermal Performance Test,
 Completed in 1999
 8.5.3.14.1, Revision 3, 2/11/97; RBCCW Heat Exchanger Thermal Performance Test,
 Completed in 2001
 8.5.3.14.1, Revision 4; RBCCW Heat Exchanger Thermal Performance Test, Completed in
 2003
 8.5.3.14.1, Revision 4; RBCCW Heat Exchanger Thermal Performance Test, Completed in
 2005
 3.M.4-85, Revision 16; Station Diving Procedure For Underwater Work And Inspections
 EN-IS-124, Revision 2, 6/1/06; Video Inspection Of E SW Pump
 8.5.3.14, Revision 26, 10/5/06; SSW Flow Rate Operability Test

- 2.2.32, Revision 74, 9/4/06; Weekly RBCCW/TBCCW Heat Exchanger Differential Pressure Evaluation
- 3.M.3-5.8, 4KV Bus Startup Transformer PT Fuse Drawer and Bus Duct Maintenance and Inspection, Rev. 4
- 3.M.1-14, General Maintenance Procedure for Heavy Load Handling Ops., Rev. 17
- 3.M.5-2, Main Stack Inspection Guideline, Rev. 8, completed 06/10/04
- 3.M.5-2, Main Stack Inspection Guideline, Rev. 9
- 3.M.2-5.3, LPRM Calibration, Rev. 19
- 3.M.7-5, Lifting Equipment, Rev. 13
- 8.M.1-1, IRM Functional/Calibration Check, Rev. 51
- 8.M.1-3, APRM Functional (Scram Clamp Normal), Rev. 54

Miscellaneous Documents

- Pilgrim Plant Programs Used Oil Analysis Guidelines, dated 09-22-2006
- CR-PNPLO-2006-00184, Snapshot Self-Assessment Results of 3rd Interval ISI Program for TS 4.6.6 requirement, 8/3/06
- Pilgrim Nuclear Power Station Risk-Informed Inservice Inspection (RI-ISI) Periodic Update Review, Revision B, 8/3/04
- Pilgrim Risk-Informed Inservice Inspection Application To Class I Piping Bases Document; Report PNPS-02Q-402, Revision 0, 9/10/01
- PR 96.0396
- PR 99.1296, 6/8/99
- PR 99.9211, 6/9/99
- PR 99.9247, 5/99
- MR #19800428, 6/7/99
- PNPS Visual Examination VT-3 Data Sheet, VT-6-05001
- PNPS Visual Examination VT-3 Data Sheet, VT-6-05004
- M-547, Revision 13, "Lubricants and Diesel Fuel Oil Specification"
- M-591, Revision 7, "GL 89-13 SSW & RBCCW Heat Exchanger Maintenance & Testing"
- ER 05107476, Revision 0, 4/30/05; Perform Engineering Evaluation of FW Hanger H6-1-SS3 and H6-1-SS5 to resolve CR-PNP-2005-01982
- Main Stack Inspection Report, 5/8/06
- Status of Pilgrim Isometric (a)(2) Review, 10/25/06

CRs (all CR-PNP-)

1989-02233	2002-10031	2005-01839
1996-00396	2002-10639	2005-01902
1996-09593	2002-11632	2005-01982
1997-01787	2002-09283	2005-04706
1997-09135	2002-09306	2006-00772
1999-01467	2003-02976	2006-00773
2000-09228	2003-03237	2006-01914
2001-02275	2003-04416	2006-03159
2001-04048	2004-00391	2006-03291
2001-04171	2004-01768	2006-03488*
2001-04779	2005-01107	2006-03689*
2001-09403		

*As a result of this inspection

Maintenance Requests/Work Orders

Preventive Maintenance Tasks:

P630025
P003564
P9400207
P9400208
02107756
04105897
04105898
04105899
05118867
06105115
06112864

System Health Reports

Heat Exchanger Component-System Health Report, 1st Quarter 2006
RBCCW System Health Report, 2nd Quarter 2006
ISI Program Health Report for 1st Quarter 2005 to 4th Quarter 2006
Fire Protection System Health Report, 3rd Quarter 2005 to 2nd Quarter 2006
Salt Service Water System Report - 4th Quarter 2004 to 2nd Quarter 2006

Miscellaneous

HPCI System Performance Monitoring Plan
LO-PNP-2005-00025-CA-0004, PNPS Flow Accelerated Corrosion Program Focused Self-Assessment Report, January 2005
RCIC Walkdown Checklist, Rev. 1
RFO 14 FAC Inspection Summary

RFO 15 FAC Inspection Data
Specification M300, Rev. 104
SBGT System Walkdown Checklist, 8/23/06
SDBD-48, Design Basis Document for Standby Gas Treatment System, Rev. 1
Standby Gas Treatment System Health Report, 2nd Quarter 2006
Summary of PNPS RFO 15 FAC Program Inspection Activities, May 2005
PNPS-RPT-05-001, Rev. 0, Fourth Ten-Year Inspection Interval In-service Inspection (ISI)
Program Plan
PDCR 79-41, Rev. 0, Feedwater Sparger Replacement and Nozzle Clad Removal
Quality Assurance Audit Report QA-08-2005-PNP-01, dated 5/25/2005
In-service Inspection Summary Report for Refuel Outage 14 at Pilgrim Nuclear Power Station,
April 2003, dated 7/3/2003
PR01.9816.00
Walkdown List for AMRM-30

NRC Documents

NRC Regulatory Guide 1.147
NRC Information Notice 89-79: Degraded Coatings And Corrosion Of Steel Containment
Vessels
NRC Information Notice 89-79, Supplement 1: Degraded Coatings And Corrosion Of Steel
Containment Vessels
NRC Information Notice 97-10: Liner Plate Corrosion In Concrete Containments
NRC Generic Letter 98-04; Potential for Degradation of the Emergency Core Cooling System
and the Containment Spray System After a Loss-Of-Cooling Accident Because of
Construction and Protective Coating Deficiencies and Foreign Material in Containment
NRC Information Notice 86-99, Supplement 1: Degradation Of Steel Containments
Generic Letter 89-13, "Safety Related Service Water and Heat Exchanger Problems"
Generic Letter 88-14, "Instrument Air System Problems"
NUREG-1522, Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures,
6/95
NUREG-6706, Capacity of Steel and Concrete Containment Vessels With Corrosion Damage,
2/01
NUREG-0661, Safety Evaluation Report Mark I Containment Long-Term Program, 7/80

Entergy Letters/Correspondence

License Renewal Application Amendment 1, 5/11/06
License Renewal Application Amendment 2, 6/7/06
License Renewal Application Amendment 3, 7/5/06
License Renewal Application Amendment 8, 9/13/06
License Renewal Application Amendment 10, 12/10/06
Entergy Ltr 2.04.027, 4/14/04; Request for Amendment to the Technical Specifications to
provide a One-time Integrated Leak Rate Test (ILRT) Interval Extension

Industry Documents

EPRI Technical Report 1010639, Revision 4, Appendix C, "Lubricating Oil and Fuel Oil"
ASTM D-975-1981, "Standard Specification for Diesel Fuel Oils"
ANSI/ISA S7.0.0.1-1996, "Quality Standard for Instrument Air"
American Society Mechanical Engineers (ASME) Boiler Pressure Vessel Code Section XI, 1998
Edition with 2000 Addenda
BWRVIP-75-A: BWR Vessel and Internals Project Technical Basis for Revisions to Generic
Letter 88-01 Inspection Schedules
NEI 95-10, Industry Guidelines for Implementing the Requirements of 10 CFR 54 - The License
Renewal Rule, Rev. 6

LIST OF ACRONYMS

ADAMS	Agency-wide Documents Access and Management System
ASME	American Society Mechanical Engineers
PARS	Publicly Available Records
GALL	Generic Aging Lessons Learned Report
BWR	Boiling Water Reactor
FAC	Flow Accelerated Corrosion
PNPS	Pilgrim Nuclear Power Station
HPCI	High Pressure Coolant Injection
RCIC	Reactor Coolant Isolation Cooling
SBGT	Standby Gas Treatment

From: Maitri Banerjee
To: graham.b.wallis@dartmouth.edu; jdsieber@aol.com; mvbonaca@snet.net; Otto Maynard; wjshack@anl.gov
Date: 03/07/2007 2:26:20 PM
Subject: Pilgrim License Renewal Subcommittee Briefing

The License Renewal Subcommittee is scheduled to be briefed on 4/4/06 at 10:30 AM on the subject, and then will brief the Full Committee in the afternoon.

I have compiled background materials including the LR application, RAI responses that amended the application, staff's audit report on the aging management program, and the draft SE with open items. I have two CDs that I will distribute to the members during the March meeting, and a hard copy will also be mailed during the same time frame. Please look for the hard copies as you return to your home office.

Please let me know if you experience any problem with accessing the documents.

Some of the Subcommittee members attended a lunch time information gathering discussion with the staff on 3/7/07. There are four open items in the SE as discussed below. The first two open items were discussed. The open item no. 1 could be a show stopper (at least that's what NRR management is communicating). Because of the evolving nature of this issue, the documents that I am sending to you do not discuss this issue in much detail.

1. Water in the torus room - Inspectors found water on the torus room floor after the NRR audit team left the site. The torus room basemat is 8 feet thick concrete, poured in vertical sections. Bottom of the floor mat is at minus 25 feet while top of the water table is at about minus 1 feet. Licensee claims ground water is seeping thru the permeable concrete. The staff is wondering if the seepage is thru any cracks in the basemat. We do not know yet how much water was found, and the status of the licensee's settlement monitoring program and the water removal system. Staff told us that licensee has done some chemical analyses and does not think water is aggressive (high pH), and also that water was found in this area during the 1990s. The staff expects to get more information from the licensee by the end of next week. The LR Subcommittee Chairman told the staff that without a resolution path having a Subcommittee meeting on 4/4/07 may not be a worthwhile endeavor, as the ACRS does not want to have a second Subcommittee meeting on the subject.

2. Neutron fluence - 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 and associated TLAAs. The staff believes there is a resolution path. The licensee was asked to determine the bounding fluence value by back-calculating from the TLAAs. The licensee was asked to install dosimetry to obtain reactor fluence numbers before entering the LR period.

3. Drywell liner integrity issue - Following the Oyster Creek drywell corrosion issue, the staff is also looking at Pilgrim's program, data and history due to similarity of design (Mark 1). The licensee claims a clean slate, but a recent audit done by the staff identified an inoperable flow switch in the bellows rupture drain. Also, monitoring of other drains has been inconclusive. The staff is asking the licensee to address the impact of all these findings on the aging management of corrosion in the inaccessible areas of the drywell liner, including in the sand pocket area.

4. The licensee needs to address how to manage the aging effect of inaccessible fire barrier penetration seals.

Please let me know if you have any questions.

Maitri

CC: Cayetano Santos

Mail Envelope Properties (45EF11DC.3CE : 12 : 35638)

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