Entergy

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May 3, 2010

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

SUBJECT: Response to Request for Additional Information – One Time Extension to ILRT – ME2122

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

References:
1. License Amendment Request for a One-Time Extension to the Integrated Leak Rate Test Interval, dated August 25, 2009 (ML092380646)
2. NRC e-mail dated February 3, 2010, Palisades – One Time Extension to the ILRT RAIs – ME2122

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (ENO) submitted a license amendment request (Reference 1) for a one-time extension to the ten-year frequency for the next Palisades Nuclear Plant containment type A integrated leak rate test that is required by Technical Specification 5.5.14. ENO received an electronic request for additional information (RAI) from the Nuclear Regulatory Commission (NRC) (Reference 2). ENO and the NRC held a conference call on February 23, 2010, to clarify the RAI.

Attached is the ENO response to the RAI.

A copy of this request has been provided to the designated representative of the State of Michigan.

This letter contains no new or revised commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 3, 2010.

Sincerely,

cjs/jlk

Attachment: 1. RAI Response on One Time Extension to ILRT Interval LAR

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

ATTACHMENT 1
RAI RESPONSE ON ONE TIME EXTENSION TO ILRT INTERVAL LAR

Request for additional information received by electronic mail February 3, 2010

Nuclear Regulatory Commission (NRC) Request

1. *Section 4.1 of Attachment 1 to the LAR* [license amendment request] *stated that the second, third, and fourth post-operational ILRT* [integrated leak rate test] *tests, resulted in the combined calculated leakage plus the adjusted measured penetration leakage exceeding the acceptance criteria. Please provide a brief description of the reasons for exceeding the acceptance criteria, fixes made to bring the leakage to within compliance, and the performance of the items to which fixes were made during subsequent leakage tests.*

Entergy Nuclear Operations, Inc (ENO) Response

1. The following paraphrased excerpts from the Palisades Nuclear Plant (PNP) Final Safety Analysis Report (FSAR) describe the reasons for exceeding the acceptance criteria during the second, third, and fourth post-operational ILRT tests. Also provided is information from the April 11, 1986, NRC Inspection Report identifying a violation (255/86005-04) with respect to the methodology that was being used at PNP at that time that did not incorporate the results of local leak rate tests (LLRT) into the Type A test results. As documented in the inspection report, the second, third, and fourth post-operational as-found ILRTs failed due to incorporating the LLRT adjustments into the Type A test results. The fix that is described further at the end of this response addressed the violation and ILRT failures by correcting the methodology and incorporating the LLRT results into the ILRT results. During the fourth post-operational ILRT, in 1986, and subsequent ILRTs the methodology was applied correctly.

   FSAR Section 5.8.8.1.1 Historical Summary:

   Second Postoperational ILRT

   A second postoperational Type A test was completed on March 28, 1978. This was a reduced pressure test (28 psig). This test was conducted within the general guidelines of the Technical Specifications (TS), 10 CFR 50 Appendix J, and ANSI N45.4.

   On February 11, 1975, the TS were amended and the temperature correction factor was eliminated. The basis for this change was that such a correction is not required by 10 CFR 50 Appendix J, which was issued several years after the original TS. This correction factor was not applied to this test.

   During containment pressurization, a leak was found on the 48-inch containment purge air exhaust penetration control valve grease fitting. The leak was measured and recorded and the grease fitting was replaced to correct the problem. The penetration was then leak tested again. The

containment leakage rate after repair of the penetration was well within the acceptance criteria. In fact, the calculations showed a net inflow into the containment. The combined calculated containment leak plus the adjusted measured penetration leak were above the acceptance criteria.

The negative leakage (net inflow) was attributed to instrument error, which for this test was found to be approximately 40% of the acceptance criteria value. Since this magnitude of error can significantly impact future test results, Consumers Power Company (CPCo) [former owner of PNP] performed a review of test monitoring equipment and procedures. As a result of this review, new instrumentation requirements were established for the ILRT, using ANS 56.8-1981, as a guide.

No further mention was made in subsequent ILRT reports on leakage from the control valve grease fitting.

Third Postoperational ILRT

The third postoperational Type A test was concluded on November 18, 1981. The test was a reduced pressure test of 28 psig. The test was conducted within the general guidelines of the TS, 10 CFR 50 Appendix J, and ANSI N45.4.

The containment leakage rate was within the acceptance criteria, but the April 1986 NRC violation 255\86-005-04 identified that CPCo had not complied with a requirement that LLRT adjustments be added to the ILRT result. This failure to calculate the as-found penalty resulted in the as-found leakage rate exceeding the acceptance criteria in both this 1981 test and the 1978 test.

Fourth Postoperational ILRT

The fourth postoperational Type A test was conducted in January 1986. The test was a reduced pressure test (28 psig). The test was conducted within the general guidelines of the TS, 10 CFR 50 Appendix J, and ANSI N45.4.

The as-found leakage rate exceeded the acceptance criteria when adjustments were added to the calculated leak rate.

As noted above, an April 11, 1986, NRC Inspection Report to CPCo provided the results of an NRC inspection that described violation 255/86-005-04. The violation concerned the methodology used at PNP. Containment isolation valve leak testing and isolation valve repair prior to Type A tests had been done, in 1978 and 1981, without adding the LLRT differences to the final ILRT results to determine the subsequent retest schedule. Adding in the LLRT results led to PNP failing both the 1978 and 1981 Type A tests in the as-found condition.

CPCo responded to the violation, in a May 9, 1986, letter to the NRC, and identified the corrective actions taken to correct the methodology. Specifically, the ILRT procedure was revised to address adding LLRT differences to the ILRT results to determine the subsequent retest schedule.

The 1986 containment ILRT failed its as-found leakage rate criteria due to the addition of repair and adjustment penalties. The as-found failure was largely due to leakage through containment penetration number 40, reactor coolant sample line, and penetration number 69, clean waste receiver tank (CWRT) pump suction line.

The penetration 40 reactor coolant sample line isolation valves were replaced during the refueling outage. Operational changes have been implemented that have reduced the potential for valve degradation as evidenced by satisfactory leakage test results for the past nine years.

The penetration 69 CWRT pump suction line isolation valves were repaired in 1986. Penetration 69 isolation valve performance in the last 15 years has been satisfactory except for an occasion in 2004. Following a leak rate test during repair activities, resin residue was found on the valve seat of one of the isolation valves.

NRC Request

2. *Section 4.1 of Attachment 1 to the LAR stated that the current total penetration leakage for Type B and Type C tests on a maximum path basis is less than 11% of the leakage allowed for containment integrity. Provide the as-found minimum pathway combined Type B & C total leakage values for the last refueling outage when a Type A test was performed and the subsequent combined as-found Type B & C test values since then. Provide this data in terms of percentage of leakage allowed (0.6L_a) and indicate if the L_a* [L_a is the maximum allowable leakage rate at the calculated peak containment internal pressure related to the design basis accident] *value has changed during this period.*

ENO Response

2. The last Type A test was performed in 2001. The minimum pathway combined Type B & C total leakage value from the 2001 refueling outage is provided in row 1 of Table A below. The data is also provided in terms of percentage of leakage allowed (0.6L_a).

   The subsequent combined as-found Type B & C test values since the 2001 refueling outage are provided in rows 2-15 of Table A below. The data is also provided in terms of percentage of leakage allowed (0.6L_a). The L_a value of

148,465 standard cubic centimeters per minute (sccm) has not changed during the period since the last Type A test was performed.

Table A – Combined B & C leakage values since last ILRT				
Date	As-Found Leakage (sccm)	L_a (sccm)	Percentage (As-Found/ L_a x100)	Percent $0.6L_a$ (As-Found/$0.6L_a$ x100)
1. 05/03/2001	12,700.6	148,465	8.55	14.26
2. 12/27/2001	33,077.7*	148,465	22.28	37.13
3. 04/15/2002	12,989.9	148,465	8.75	14.58
4. 02/06/2003	13,010.8	148,465	8.76	14.61
5. 04/16/2003	14,118.7	148,465	9.51	15.85
6. 12/16/2003	14,488.4	148,465	9.76	16.26
7. 11/11/2004	47,840**	148,465	32.22	53.71
8. 11/15/2004	14,928.7	148,465	10.06	16.76
9. 05/04/2006	14,074.7	148,465	9.48	15.80
10. 05/15/2006	15,156.3	148,465	10.21	17.01
11. 01/04/2007	14,840	148,465	10.0	16.66
12. 10/13/2007	16,444.8	148,465	11.08	18.46
13. 11/17/2008	14,669.3	148,465	9.88	16.46
14. 04/18/2009	15,768.72	148,465	10.62	17.70
15. 04/27/2009	15,466.96	148,465	10.42	17.36

* The largest contributor to the as-found leak rate was penetration MZ-66, "ILRT Instrument Line," which had a leak rate of 22,142 cc/min. Subsequent to the initial test the penetration was blown down and retested. The as-left leak rate for MZ-66 was 2,010.5 cc/min, which was acceptable.

** Leakage through control valves CV-1044 and CV-1045, clean waste receiver tank pump suction, in penetration MZ-69, exceeded the administrative limit and the valves were placed on a 30-month test frequency in accordance with the local leak rate test (LLRT) program.

NRC Request

3. *The Type A Containment Integrated Leak Rate Test (ILRT), the Type B and Type C Local Leak Rate Tests (LLRTs), and Containment In-Service Inspection (CISI) program collectively ensure leak-tight integrity and structural integrity of the containment. Please provide the following for the PNP:*

a. *The as-found and as-left Type A test results and their comparison with the allowable leakage rate specified in the plant Technical Specifications.*

ENO Response

3.a. Tables B and C provide the Type A test results. Following the tables are discussions of the Type A test results. Table B provides the as-found test result as compared to the Technical Specifications (TS) acceptance criteria from the time that the Type A tests were performed. Table C provides the as-found and as-left Type A post-operational test results and the comparisons to the TS allowable leakage rate at the time the Type A tests were performed.

Table B – 1974 - 1981 ILRT Results			
Test Date	Test Results (wt %/day)	TS Acceptance Criteria (wt %/day)	Comments
April 1974	0.0342	0.0514	1
March 1978	0.09242	0.0559	2
November 1981	0.0349	0.0713	3

1. Test results were calculated for a test pressure of 28 psig. A TS acceptance criterion was 0.0514 wt %/day with an Appendix J acceptance criteria of 0.0386 wt %/day.
2. Test results were -0.00708 wt %/day. However a leaking penetration required adjustment to 0.09242 wt %/day.
3. Appendix J allowable leakage rate was 0.0535 wt %/day.

Table C – 1986 – 2001 ILRT Results					
Test Date	As-Found Test Results (wt %/day)	As-Left Test Results (wt %/day)	TS Acceptance Criteria (wt %/day)	Appendix J Acceptance Criteria (wt %/day)	Comments
January 1986	0.1061	0.0290	0.1	0.075	Reduced pressure test
November 1988	0.0408	0.02617	0.1	0.075	Reduced pressure test
February 1991	*	0.070439	0.1	0.075	Full pressure test
May 2001	0.0140	0.0122	0.1	0.075	Full pressure test

* The 1991 test was performed after creation and repair of a containment opening for the steam generator replacements. Therefore, there was no as-found test result.

April 30, 1974 – The calculated leak rate at reduced test pressure of 28 psig was 0.0342 weight percent (wt %)/day of contained air. The maximum allowable TS leak rate (L_t) for this reduced pressure test was 0.0514 wt %/day. In accordance with the then existing version of Appendix J, the allowable operational leak rate was 75% of the maximum allowable in order to allow for possible degradation. Therefore, the allowable operational leak rate per Appendix J was 0.0386 wt %/day.

March 28, 1978 – As-found leakage rate at 28 psig of -0.00708 wt %/day with a 95% upper confidence limit of 0.00195 wt %/day and a TS limit of 0.0559 wt %/day. The negative leakage rate was attributed to instrument error. During the test, a containment leak was discovered on the containment purge air exhaust penetration fitting. This fitting was replaced. The leak rate from this fitting was added to the test resultant leak rate, of -0.00708 wt %/day, providing an as-found leakage was 0.09242 wt %/day.

November 18, 1981 – Calculated nominal leakage rate of 0.0326 wt %/day with a one sided 95% upper confidence limit of 0.0349 wt %/day. Assuming the containment pressure of 28 psig, the maximum allowable leakage rate was 0.0713 wt %/day. The required leakage rate could not exceed 75% of the allowable leakage rate or 0.0535 wt %/day.

January 1986 – A reduced test pressure (P_t) of 28.25 psig was recorded at the end of the 30-hour hold test. At P_t, the measured containment leak rate (L_{tm}) for the hold test was 0.0157 wt %/day with a 95% upper confidence limit of 0.0187 wt %/day. After upward adjustment to accident pressure (P_a), the measured leak rate (L_{am}) was 0.0262 wt %/day at a 95% upper confidence limit. The calculated as-left total containment leak rate was 0.0290 wt %/day at the 95% upper confidence limit, following Type C test additions for systems not in their accident status during the conduct of the Type A test and after compensating for a 2% increase in the pressurizer level (equivalent to 0.0017 wt %/day).

At the accident pressure (P_a) of 55 psig, the maximum acceptable leak rate for a Type A test is 0.075 wt %/day (0.75 L_a) per 10 CFR50 Appendix J, Section III.A.5.b.1. This includes corrective additions to account for omitted systems and containment free volume changes.

Following penalty additions for outage repairs resulting in improvements in Type B & C leak rates, the as-found Type A leak rate was 0.1061 wt %/day at the 95% upper confidence limit, which did not meet the acceptance criteria of 0.075 wt %/day(0.75 L_a). Two-thirds of the total penalty assessed was based on a conservative minimum pathway determination for penetration numbers 40 and 69, necessitated through replacement and/or repair of double isolation valves.

November 1988 – The measured containment leak rate (L_{tm}) at the reduced test pressure of 28.66 psig was 0.01651 wt %/day with a 95% upper confidence limit

of 0.01758 wt %/day. Adjusting the reduced pressure leak rate upward to P_a results in a measured leak rate (L_{am}) of 0.0231 wt %/day, with a 95% confidence limit of 0.0246 wt %/day. The addition of Type B & C penalties and compensating for a 1% increase in pressurizer level, results in a calculated as-left total containment leak rate of 0.02617 wt %/day at the 95% upper confidence limit adjusted to P_a.

The as-found Type A leak rate was 0.02915 wt %/day at the 95% upper confidence limit or 0.0408 wt %/day (95% upper confidence limit) corrected upward for accident pressure (P_a) of 55 psig. This value is within the acceptance criteria of 0.075 wt %/day (0.75 L_a).

February 1991 – The measured containment leak rate (L_{am}) for the hold test at 55.61 psig was 0.02473 wt %/day and an upper 95% confidence level was 0.0700838 wt %/day. The addition of the Type B & C penalties resulted in a calculated as-left total containment leak rate of 0.070439 wt %/day. Referencing 10CFR50, Appendix J, Section III.A.5.b.1, the maximum acceptable leak rate for a Type A test at the P_a of 55 psig is 0.075 wt %/day (0.75 L_a), including the corrective additions to account for omitted systems and the containment free volume changes.

The as-found condition is the condition of the containment at the beginning of the outage prior to any repairs or adjustments to the containment boundary. This is normally calculated by reviewing the summary of the LLRTs and calculating the amount of leakage rate improvements due to repairs or adjustments using minimum pathway methodology. This assumes that no major changes to the containment structure were made, but that all leakage improvements were due to penetration repairs or adjustments. However, during the 1991 outage, a hole was cut through the primary containment structure in order to allow replacement of the steam generators. Thus, no correlation could be established between the pre- and post- modification leakage rates. Therefore, this ILRT was considered to be a pre-operational test to show that the repairs to the containment adequately met the TS leakage requirements, rather than the performance of a periodic ILRT.

May 2001 – The pressure across the containment boundary was 53.524 psig (outside barometric pressure of 14.558 psia). An acceptable mass point leakage rate at the 95% upper confidence limit of 0.0100 wt %/day was determined not including the leakage rate corrections. Total leakage rate corrections (Type B & C LLRT penalties and water volume corrections) were determined to be 0.0022 wt %/day. The as-left 95% upper confidence limit leakage rate was determined to be 0.0122 wt %/day. The maximum allowable leakage rate (L_a) for the containment was 0.1 wt %/day with a test acceptance of 0.075 wt %/day (0.75 L_a).

The as-found 95% upper confidence limit leakage rate was 0.0140 wt %/day, which included a leakage savings of 0.0018 wt %/day.

NRC Request

4. *The staff noted that the PNP will be entering into the period of extended operation (PEO) on March 24, 2011. By Technical Specification (TS) 5.5.14, PNP is required to perform its ILRT test by May 3, 2011, which is approximately six months after 1R21. Based on the licensee's relief request, PNP has requested to extend its ILRT test interval to August 3, 2012. However, in NUREG-1871, "SER Related to the License Renewal of Palisades Nuclear Plant," January 2007, on page 3-22, the staff states that the plant-specific operating experience revealed some instances where the Containment Inservice Inspection Program had been instrumental in discovering material degradation. Containment degradation included liner plate corrosion, unacceptable tendon liftoff values, tendon gallery corrosion, tendon grease leakage, the moisture barrier not in place, and tendon sheath water intrusion.*

   Based on the information above, the staff requests that the licensee:

a. *Justify its basis for not conducting the ILRT during 1R21, which is scheduled for October 2010.*

ENO Response

4.a As indicated above on page 3-22 of NUREG-1871, the NRC staff stated that the plant-specific operating experience revealed some instances where the CISI program had been instrumental in discovering material degradation. Containment degradation included liner plate corrosion, unacceptable tendon liftoff values, tendon gallery corrosion, tendon grease leakage, the moisture barrier not in place, and tendon sheath water intrusion.

Also, in NUREG-1871, page 3-22, the NRC stated:

> The s*taff's review of the operating experience related to the CISI program revealed no significant failures of the containment shell concrete, post-tensioning system, and steel pressure-retaining elements due to degradation. The operating experience was typical of the types of minor material degradation found on the interior, the exterior, and with the tendon system for a prestressed concrete containment. The staff did not identify any operating experience that would require any modification to the program.*
>
> *The staff reviewed the operating experience provided in the LRA* [License Renewal Application] *and interviewed the applicant's technical personnel*

to confirm that the plant-specific operating experience revealed no degradation not bounded by industry experience.

Furthermore, review of additional industry operating experience related to the containment leakage testing program led the NRC staff to state the following on page 3-24 of NUREG-1871:

> *Site operating experience shows that no significant problems have been found during periodic Type A tests. This confirms that the local leakage rate testing program (in conjunction with periodic containment examinations) has always detected developing deterioration before it could result in a loss of containment leak-tight integrity (defined by overall leakage exceeding L_a).*

The data provided in response to RAI item 2 above shows that the PNP containment is leak-tight with respect to the overall leakage exceeding L_a. The response to RAI item 4.b below indicates that the PNP CISI program has continued to reveal that there are no significant failures of the containment shell concrete, post-tensioning system, or steel pressure-retaining elements due to degradation.

Justification for not conducting the ILRT during 1R21 is in part based on the same conclusions described above that the NRC made in NUREG-1871. Further justification is in the NUREG-1871 discussion of the NRC acceptance of the program elements of the CISI and Containment Leakage Testing Program and the conclusion that the effects of aging will be adequately managed. This justification is aligned with the justification in the ENO statement on the history of the programs' inspections in the August 25, 2009, LAR that indicated "ENO is proposing this [license] revision based on the good containment leakage rate history and containment visual examination history at PNP, and because there is no substantial increase in risk associated with extending the inspection interval 15 months, ..."

NRC Request

4.b. *Examination results of IWE* [American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWE (metallic liner)] *and IWL* [ASME B&PV Code, Section XI, Subsection IWL (concrete components)] *programs, and any corrective /preventive actions, acceptance criteria, and monitoring and trending were taken.*

ENO Response

4.b. The following information provides summaries of the PNP IWE examination results of the containment metal liner completed during refueling outage 1R18

(2006) and 1R20 (2009). Also provided are the PNP IWL examination results of the containment concrete visual inspections completed in 2000 and 2005 and tendon inspections completed in 2002 and 2008. Corrective actions identified by these inspections are provided with the information.

IWE Examinations

Refueling Outage 1R18 (2006) Containment In-Service Inspection-Metal Liner

Examinations performed for the containment liner plate per TS Surveillance Procedure RT-142, "Containment Inservice Inspection-Metal Liner," identified several small areas that were recorded as indications in the non-destructive examination (NDE) reports during refueling outage 1R18. The indications were documented in the PNP corrective action program. In a letter provided by the NDE inspectors these visual observation indications were categorized as surface corrosion that was not excessive. The indications were validated by ultrasonic testing in several areas that were determined to be representative of all corroded areas in containment. The minimum thickness reported for these areas was 0.234 (nominal 0.250) inches and 0.485 (nominal 0.500) inches. The RT-142 examinations also identified a small area of missing moisture barrier, which was documented in the PNP corrective action program. This area of the moisture barrier was subsequently replaced and successfully re-inspected.

All reported visual observations were considered cosmetic with no areas of suspect damage or deterioration, which would impact the structural integrity or leak tightness of the containment liner. The RT-142 examination of the containment liner plate was successfully completed in 1R18 and was "Accepted by Examination" in accordance with applicable requirements.

Refueling Outage 1R20 (2009) Containment In-Service Inspection-Metal Liner

Examinations performed for RT-142 identified only one recordable condition that was not previously evaluated. The recordable condition, documented in the PNP corrective action program, was excessive boric acid on penetration number 68. After the boric acid was removed, a subsequent inspection was performed and it found penetration number 68 to be acceptable. All the visual observations noted by this inspection were cosmetic with no areas of suspect damage or deterioration, which would impact the structural integrity or leak tightness of the containment liner.

In addition to the containment metal liner inspection performed during refueling outage 1R20, a general visual inspection of all accessible exterior surfaces of the containment was performed per TS Surveillance Procedure RT-203, "Containment Visual Inspection." The completion of the general visual inspection

of the exterior surfaces of the containment resulted in finding no structural problems that could affect the containment structure leakage integrity. Minor observations were identified and recorded in the inspection report to be monitored and trended in future engineering program inspections.

IWL Examinations

A. IWL (Concrete)

Visual examinations were performed, in 2000, under TS Surveillance Procedure FT-7, "Containment Visual Inspection" to satisfy TS 4.5 and 6.6 requirements. During examination of auxiliary building room 232, purification filter room, oil was observed at the tendon buttress. This oil was evidence of grease leakage from one or more tendons at the buttress. During examination of the containment from the tendon access tunnel, grease leakage was discovered at tendons V-176 and V-306; concrete "pop-outs" exposing near surface rebar were discovered near tendons V-128, V-126, V-82 and V-86; a concrete "pop-out," which did not expose rebar, was discovered near tendon location V-208. Tendons were not found installed at locations V-142 and V-208. These observations were documented in the PNP corrective action program. The missing tendons are documented in the PNP FSAR Section 5.8.2. The concrete "pop-outs" discovered in the tendon access tunnel were considered to involve the concrete surface and did not affect the containment basemat. Historical information documented in the PNP corrective action program has indicated that grease leakage has not resulted in tendon wire corrosion.

Visual examinations were performed, during June 2005, under FT-7 to satisfy TS administrative requirements 5.5.5 and 5.6.7. During the examination various minor recordable indications were observed. Examples of these indications included: 1) tendon grease at various tendon buttresses, 2) actual grease leakage was observed at tendon caps in the tendon tunnel and on the containment dome, and 3) concrete "pop-outs," "spalls," "cracks" and indications described in the visual examination procedure. These observations were entered into the PNP corrective action program and evaluated. All of the observations of degradation identified during the performance of FT-7 were found to be minor with no operability concerns. It was determined that all of the indications affected only the outer portions of the concrete structure and the indications were cosmetic.

B. IWL (Tendons)

30-year tendon surveillance

The 30-year tendon surveillance activities were completed in January 2003. The former plant operator Nuclear Management Company, LLC submitted the results

of the surveillance in a letter dated April 7, 2003. A summary of the examination and inspection results and corrective actions from this surveillance follow.

The following tendons exceeded the acceptance criteria for grease replacement:

- Dome Tendon D1-38, 18.3 gallons or 32 percent
- Vertical Tendon V-16, 11.3 gallons or 14.4 percent
- Vertical Tendon V-30, 9.9 gallons or 12.6 percent
- Vertical Tendon V-116, 8.8 gallons or 11.2 percent
- Vertical Tendon V-330, 8.2 gallons or 10.5 percent

Each of these tendons met the criteria for force measurement, anchorage hardware and surrounding concrete. Tendon wire surfaces were fully covered with a protective grease coating. All locations were refilled by injecting new grease. These tendons were determined to be fully operable.

Grease leakage was discovered at the main gaskets for vertical tendons V-98, V-132, V-134, V-150, V-154, V-178, V-218, V-166, and V-186. All leaks were from the top or "shop" end on the containment dome. It was determined that filling of the grease cans, during the steam generator replacement project, which occurred during a cold weather period in 1991, and subsequent heating, expanded the grease and pushed it by the main gasket. The main gaskets were replaced and grease leakage from the subject cans stopped as part of the tendon surveillance. The quantity of grease replacement was sufficient to cover tendon end-anchorage hardware but an air pocket was left in the upper portion of the can to allow grease expansion and contraction. The protective grease layer on the tendons was not compromised by the small amount of leakage.

Surveillance activities discovered three missing button heads on vertical tendon V-30, "field end." One of the missing button ends was previously recorded during plant construction. The cause of the two additional failed button heads appeared to be fabrication flaws that occurred during the button heading process as evidenced by the lack of button heads in the removed grease. There was no visible sign of deterioration at the end of the individual tendon wires. Vertical tendon V-30 met all the other applicable tests and inspection criteria.

Surveillance activities discovered a single protruding wire at the "shop end" of dome tendon D3-22. It was determined that the protruding wire in dome tendon D3-22 was due to a break somewhere along the length of the wire as evidenced by the button head being in place at the "field end." Efforts to remove the wire were unsuccessful, making it impossible to determine the cause of failure. There was no visible sign of deterioration at the ends of the wires. Tendon force measurement testing was not performed on dome tendon D3-22 due to obstructions. However, all other test and inspection acceptance criteria were met for grease coating, sampling and loss, inspection for water, anchorage corrosion, and concrete inspection.

Water infiltration has been documented during previous tendon surveillances at PNP. The cause of water infiltration has been traced to degraded grease can gaskets and migration through concrete cold joints and tendon sheathing. Surveillance activities discovered 20 ounces of free water at the "shop end" of dome tendon D1-38. Additionally, the grease sample testing for this tendon indicated chemically combined water at 11 percent at the "shop end," only. To fully determine tendon condition, tendon force measure tests and visual exams were performed. Force measurements tests were satisfactory and inspections did not discover any degradation of anchorage components. On the basis of this information, it was concluded that the presence of free water or chemically combined water in the grease was insufficient to cause corrosion or cracking of the anchorage components. Dome tendon D1-38 was refilled by injecting new grease.

In summary, it was concluded that the 30-year containment structural integrity surveillance program had demonstrated continued containment operability and that the containment structure had not experienced abnormal degradation related to the post-tensioning system.

35-year tendon surveillance

The 35th year tendon surveillance activities were completed in September 2008. ENO submitted the results to the NRC in a letter dated January 26, 2009. The examination and inspection results and corrective actions from that surveillance follow.

The sheathing filler (grease) samples were tested and found to have acceptable levels of water-soluble ions (chlorides, nitrates and sulfides). All neutralization numbers were above the IWL requirement and acceptable. Two tendon ends were found with water content above 10% by weight. The top end of vertical tendon V-212 was found with 14% water by weight and the southeast end of dome tendon D1-38 was found with 28% water by weight. Both tendon ends had acceptable grease coverage and corrosion inspection results. Water content values were below 10% by weight and acceptable for all other samples tested.

Two tendon ends exhibited water during removal of the grease cap; dome tendon D1-38 had 23 ounces removed and dome tendon D1-36 had one ounce removed. No water was exhibited during the inspection of any other tendons. A sample was obtained from dome tendon D1-38 and sent for pH testing. The sample returned with an acceptable pH level of 12.80. A sample was not able to be obtained from dome tendon D1-36 due to the small amount of water present.

Tendon ends were found to have acceptable corrosion levels and no cracks were found on any anchorage components. Cracks in the concrete surrounding the bearing plates were all within the allowable tolerance of < 0.010-inch.

All surveillance tendons monitored for forces during this inspection period were found to have forces greater than 95% of the corresponding predicted force.

The de-tensioned tendons were re-tensioned with acceptable elongations and acceptable force levels. All test wires removed from de-tensioned tendons were found to have acceptable corrosion levels. All tendon test wire samples had acceptable diameter, yield stress, ultimate stress and elongation results.

All tendons were re-sealed and re-greased to acceptable levels.

A comparison of as-found force levels to the original force levels was made in an effort to detect any evidence of system degradation. The loss of force since the original installation is comparable to the losses of other plants of this age and does not show any evidence of system degradation. The conclusion, based on the data gathered during the 2008 35-year containment IWL inspection, was that no abnormal degradation of the post tensioning system had occurred on the PNP containment structure.