Entergy

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PNP 2011-067

October 28, 2011

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

SUBJECT: Response to Request for Additional Information - Palisades - LAR to Extend the Containment Type A Leak Rate Test Frequency to 15 years - ME5997

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

References:
1. License Amendment Request to Extend the Containment Type A Leak Rate Test Frequency to 15 Years, dated April 6, 2011 (ADAMS Accession No. ML110970616)
2. NRC e-mail, of August 29, 2011, Request for Additional Information - Palisades - LAR to extend the Containment Type A Leak Rate Test Frequency to 15 years - ME5997 (ADAMS Accession No. ML112420181)
3. NRC e-mail, of September 2, 2011, Request for Additional Information - Palisades - LAR to extend the Containment Type A Leak Rate Test Frequency to 15 years - ME5997 (ADAMS Accession No. ML112510336)

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (ENO) submitted a license amendment request (Reference 1) to modify the Renewed Facility Operating License, Appendix A, Technical Specifications (TS), as they apply to the containment leak rate testing program in TS section 5.5.14. ENO received two electronic requests for additional information (RAI) from the Nuclear Regulatory Commission (NRC), on August 29 (Reference 2) and September 2, 2011 (Reference 3).

Attachment 1 contains the ENO response to the RAI in Reference 2.
Attachment 2 contains the ENO response to the RAI in Reference 3.

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 October 28, 2011.

Sincerely,

ajv/jlk

Attachments: 1. RAI Response Related to Risk Based Information on the Containment Type A Leak Rate Test Frequency

2. RAI Response Related to Information Needed Concerning Containment Type A Leak Rate Test Frequency

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

ATTACHMENT 1
RAI RESPONSE RELATED TO RISK BASED INFORMATION ON THE CONTAINMENT TYPE A LEAK RATE TEST FREQUENCY

Request for additional information received by electronic mail August 29, 2011

Nuclear Regulatory Commission (NRC) Request

1. *Page A-4 of the Palisades LAR indicates that the PNP* [Palisades Nuclear Plant] *Probabilistic Risk Assessment (PRA) substantially meets the ASME PRA Standard at Capability Category II or better for 83% of the applicable supporting requirements, with 88% meeting a capability category 1 or better. Please clarify if Appendix A addresses all findings from the peer review that does not meet capability category I and II, except those related to flooding. If not, please describe those findings and provide an evaluation of the impact on the ILRT extension request.*

Entergy Nuclear Operations (ENO) Response

1. Evaluations for all findings not related to flooding from the PRA peer review report that did not meet capability category I and II are provided in Appendix A, "PRA Technical Adequacy," Table A.2.3-1, of the LAR. Additionally, evaluations for several items that are classified as suggestions are provided in the table.

   On review of the final PRA peer review report it was found that the total number of findings and suggestions stated in the conclusion, as quoted in Section A.2.3 of the LAR submittal, did not match the totals in the peer review team report. In the peer review report, the total number of findings is 52 (14 are related to flooding), the total number of suggestions is 26, and there are two best practices.

   There are 40 findings that are not related to flooding presented in Appendix A, Table A.2.3-1, of the LAR. There should have been only 38 findings not related to flooding listed in the table. A preliminary peer review report classified two items, DA-D4-01 and LE-C9-01, as findings. However, these two items, which are on pages A-11 and A-21 respectively, of the LAR, were later changed to suggestions in the final report. The total number of findings not related to flooding was 38, with 14 specific to flooding (52 total). Evaluations for all 38 findings that are not related to flooding and did not meet capability category I and II are provided in Appendix A, Table A.2.3-1, of the LAR.

NRC Request

2. *Please clarify what version of the PRA Standard was used in the October 2009 Full Power Internal Events Peer Review. Regulatory Guide 1.200 endorses ASME/ANS RA-Sa-2009 as the current standard. If the peer review was not to the latest Regulatory Guide 1.200 Rev. 2, please describe any gaps from the peer reviewed PRA to this Regulatory Guide and the impact on this application.*

ENO Response

2. The October 2009 full power internal events peer review of the PRA was assessed against the requirements of Section 2 of the ASME/ANS RA-S-2008a combined PRA standard (Reference 1) and the requirements of Regulatory Guide (RG) 1.200, Revision 2 (Reference 2).

   The effective date for implementation of RG 1.200, Revision 2, was April 2010, which was after the PNP peer review that was done in October 2009. However, the PRA was assessed to the requirements of the ASME/ANS combined PRA standard and to RG 1.200, Revision 2. The PNP PRA peer review team lead performed a check of the Supporting Requirements (SRs) in the RA-Sa-2009 and RA-S-2008a versions and indicated the SRs match. A further comparison of approximately 80 random SRs from all technical elements from the two versions of the standard indicated the primary differences to be editorial. A check of the corresponding SR assessments in the PNP PRA peer review report Appendix B, "Overall Evaluation of Palisades PRA against the ASME PRA Standard Supporting Requirements," showed that they were consistent with the requirements in the later standard RA-Sa-2009. Based on the review described above, the conclusion reached is that the PNP PRA peer review was performed against RG 1.200, Revision 2, is consistent with ASME/ANS RA-Sa-2009, and no gap analysis is needed.

   References:

1) ASME/ANS RA-S-2008a, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME and the American Nuclear Society, December 2008.
2) Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," USNRC, March 2009.

NRC Request

3. *The risk contribution of pre-existing leakage for the pressurized-water reactor and boiling-water reactor representative plants in the EPRI Guidance confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an imperceptible increase in risk that is on the order of 0.2 percent and a fraction of one person roentgen equivalent man (rem) per year in increased public dose. Table 6-3 of the EPRI guidance summarizes figures-of-merit information for 59 plants that have applied for one-time ILRT interval extensions. In table 6-3, the maximum population dose change is 0.2 person-rem/yr.*

   In the PNP LAR, the change in dose risk for changing the Type A test frequency from three-per ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all accident sequences for PNP, is

1.34E+00 person-rem/yr using the EPRI guidance with the base case corrosion case. This value is greater than the acceptance criteria (per the EPRI guidance) for a "very small" change of <1.0 person-rem/yr.

PNP indicates that conservatisms were incorporated into this value. PNP indicated that the calculated increase is conservatively high based on the assignment of the L-LL release category to the intact containment case, which subsequently yields conservative estimates of the EPRI Class 3a and 3b calculated dose results. As provided in the EPRI guidance, "Despite very conservative assumptions, the submittals to date have been able to demonstrate that the revised ILRT testing interval has little impact on risk."

Using the approved EPRI Methodology, please demonstrate that PNP can meet the acceptance criteria of <1.0 person-rem/yr.

Note: The PNP LAR indicated that the change in dose risk drops to 2.55E-01 person-rem/yr when using the EPRI expert elicitation methodology. The Nuclear Regulatory Commission (NRC) staff has not accepted the EPRI expert elicitation as presented in the appendices of EPRI Report No. 1009325, Revision 2 (See SER)(ADAMS ML081140105). The NRC staff concerns with the EPRI expert elicitation are documented in an NRC letter dated April 22, 20[05].

ENO Response

3. The RAI issue stems from the choice of the Late-Low Low (L-LL) release category to represent the INTACT release category in the original risk assessment for the ILRT extension request. The choice of the L-LL release category was noted as conservative in the assessment, but was chosen since a corresponding release for the INTACT category was not available from the Severe Accident Mitigation Alternatives (SAMA) analysis that was previously performed for PNP and used as input into the ILRT risk assessment. However, to more properly reflect the population dose associated with the INTACT release category, a more detailed assessment is needed.

   Therefore, to better characterize the release associated with the INTACT release category, the same methodology used for determining the population doses for the other release categories used in the ILRT risk assessment was performed. That is, instead of conservatively assigning the population dose associated with the L-LL release category to the INTACT release category, a Modular Accident Analysis Program (MAAP) run was performed for a representative INTACT release category case to define the appropriate source terms associated with this release category, and then a MELCOR Accident Consequence Code System MACCS2 analysis was performed to determine the population dose associated with this release

category. The details of this revised evaluation are provided in Enclosure A. The end result is that the revised evaluation predicts a population dose of 2.08E+03 person-rem as representing the allowable leakage release case instead of 4.10E+04 person-rem that was conservatively used in the original assessment.

Since the change in population dose risk, using the approved EPRI methodology, uses linear relationships, in estimating the dose risk increase associated with the ILRT extension, the net result for the change in population dose risk associated with the ILRT assessment will scale approximately linearly with the population dose associated with the INTACT case. Therefore, the change from the original value of 1.34 person-rem/yr associated with the ILRT extension request will approximately reduce to (2.08E3 / 4.10E4) x 1.34 = 0.068 person-rem/yr. This is clearly below the acceptance criteria of <1.0 person-rem/yr and is more in line with the information provided by other sites for similar ILRT extension requests. The more detailed assessment showing the revised results is shown in the tables below. Reference is made to the original table numbers from the submittal to aid in the comparison and review of the revised results.

In conclusion, the change in dose risk for changing the Type A test frequency from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all accident sequences, is 6.82E-02 person-rem/yr using the approved EPRI guidance with the base case corrosion case (Table 5.6-1). The value calculated per the approved EPRI guidance is lower than the acceptance guideline of ≤1.0 person-rem/yr and is more in line with the information provided by other sites for similar ILRT extension requests.

Table 5.2-1 is revised to show the revised PNP population dose given that the INTACT release category case is based on more representative source terms. An additional column is added to show the revised population dose associated with each accident class. Changes from the original assessment are bolded.

Table 5.2-2 is revised to show the population dose as a function of accident class for the 3 in 10 year ILRT frequency. Changes from the original assessment are bolded.

Table 5.3-1 is revised to show the population dose as a function of accident class for the 1 in 10 year ILRT frequency. Changes from the original assessment are bolded.

Table 5.3-2 is revised to show the population dose as a function of accident class for the 1 in 15 year ILRT frequency. Changes from the original assessment are bolded.

Finally, Table 5.6-1 summarizes the revised results and shows the changes in dose rate, large early release frequency (LERF), and containment conditional failure probability (CCFP). Again, changes from the original assessment are bolded.

Table 5.2-1 (Revised)
PNP Population Dose
for Population Within 50 Miles

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES) - ORIGINAL ANALYSIS	PERSON-REM (0-50 MILES) - REVISED ANALYSIS
1	No Containment Failure (1 La)	4.10E+04	**2.08E+03**
2	Large Isolation Failures (Failure to Close)	6.15E+06	6.15E+06
3a	Small Isolation Failures (liner breach)	4.10E+05	**2.08E+04**
3b	Large Isolation Failures (liner breach)	4.10E+06	**2.08E+05**
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.07E+06	1.07E+06
7 LERF	Failures Induced by Phenomena (LERF)	6.15E+06	6.15E+06
8 non-LERF	Containment Bypass (non-LERF)	1.84E+06	1.84E+06
8 LERF	Containment Bypass (LERF)	6.15E+06	6.15E+06

Table 5.2-2 (Revised)
PNP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 3 in 10 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) [1]
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	No Containment Failure [2]	**2.08E+03**	8.71E-06	**1.81E-02**	8.71E-06	**1.81E-02**	--
2	Large Isolation Failures (Failure to Close)	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	--
3a	Small Isolation Failures (liner breach)	**2.08E+04**	2.41E-07	**5.01E-03**	2.41E-07	**5.01E-03**	--
3b	Large Isolation Failures (liner breach)	**2.08E+05**	6.02E-08	**1.25E-02**	6.04E-08	**1.26E-02**	**1E-4**
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A

Table 5.2-2 (Revised)
PNP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 3 in 10 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	--
7 LERF	Failures Induced by Phenomena (non-LERF)	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	--
8 non-LERF	Containment Bypass (non-LERF)	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	--
8 LERF	Containment Bypass (LERF)	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	--
CDF	**All CET end states**		2.66E-05	**2.430E+01**	2.66E-05	**2.430E+01**	**1E-4**

(1) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.
(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-1 (Revised)
PNP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 1 in 10 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	No Containment Failure (2)	**2.08E+03**	8.00E-06	**1.66E-02**	8.00E-06	**1.66E-02**	--
2	Large Isolation Failures (Failure to Close)	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	--
3a	Small Isolation Failures (liner breach)	**2.08E+04**	8.02E-07	**1.67E-02**	8.02E-07	**1.67E-02**	--
3b	Large Isolation Failures (liner breach)	**2.08E+05**	2.01E-07	**4.17E-02**	2.02E-07	**4.20E-02**	**2.8E-4**
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A

Table 5.3-1 (Revised)
PNP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 1 in 10 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) [1]
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	--
7 LERF	Failures Induced by Phenomena (non-LERF)	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	--
8 non-LERF	Containment Bypass (non-LERF)	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	--
8 LERF	Containment Bypass (LERF)	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	--
CDF	**All CET end states**		2.66E-05	**2.434E+01**	2.66E-05	**2.434E+01**	**2.8E-4**

[1] Only release classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.

[2] Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-2 (Revised)
PNP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 1 in 15 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	No Containment Failure (2)	**2.08E+03**	7.50E-06	**1.56E-02**	7.50E-06	**1.56E-02**	**--**
2	Large Isolation Failures (Failure to Close)	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	--
3a	Small Isolation Failures (liner breach)	**2.08E+04**	1.20E-06	**2.51E-02**	1.20E-06	**2.51E-02**	--
3b	Large Isolation Failures (liner breach)	**2.08E+05**	3.01E-07	**6.263E-02**	3.04E-07	**6.327E-02**	**6.4E-4**
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A

Table 5.3-2 (Revised)
PNP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 1 in 15 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	--
7 LERF	Failures Induced by Phenomena (non-LERF)	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	--
8 non-LERF	Containment Bypass (non-LERF)	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	--
8 LERF	Containment Bypass (LERF)	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	--
CDF	**All CET end states**		2.66E-05	**2.437E+01**	2.66E-05	**2.437E+01**	**6.4E-4**

(1) Only release classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.6-1 (Revised)
PNP ILRT Cases:
Base, 3 to 10, and 3 to 15 Yr Extensions
(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	**2.08E+03**	8.71E-06	**1.81E-02**	8.00E-06	**1.66E-02**	7.50E-06	**1.56E-02**
2	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	1.31E-07	8.06E-01
3a	**2.08E+04**	2.41E-07	**5.01E-03**	8.02E-07	**1.67E-02**	1.20E-06	**2.51E-02**
3b	**2.08E+05**	6.04E-08	**1.26E-02**	2.02E-07	**4.20E-02**	3.04E-07	**6.33E-02**
7 non-LERF	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	1.29E-05	1.38E+01
7 LERF	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	2.88E-07	1.77E+00
8 non-LERF	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	4.28E-06	7.89E+00
8 LERF	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	1.03E-13	6.36E-07
Total		2.66E-05	**2.430E+01**	2.66E-05	**2.434E+01**	2.66E-05	**2.437E+01**

ILRT Dose Rate from 3a and 3b		**1.76E-02**	**5.87E-02**	**8.83E-02**
Delta Total Dose Rate[1]	From 3 yr	---	**3.96E-02**	**6.82E-02**
	From 10 yr	---	---	**2.86E-02**
3b Frequency (LERF)		6.04E-08	2.02E-07	3.04E-07
Delta 3b LERF	From 3 yr	---	1.42E-07	2.44E-07
	From 10 yr	---	---	1.02E-07
CCFP %		66.37%	66.90%	67.28%
Delta CCFP %	From 3 yr	---	0.53%	0.91%
	From 10 yr	---	---	0.38%

1. The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

ATTACHMENT 1, ENCLOSURE A

TO RAI ITEM 3 RESPONSE

Summary of Revised MACCS2 Analysis for the INTACT Release Category

Purpose

This attachment summarizes the development and results of the Palisades MACCS2 intact containment core damage case for use in addressing the Palisades ILRT risk evaluation RAI#3.

Method

The MACCS2 files developed and used for the Palisades SAMA evaluation are used. The MACCS2 results from these files are used as inputs to the ILRT risk evaluation. A new source term case is added to the MACCS2 ATMOS to model an intact containment core damage release. Where appropriate, the intact containment release category characteristics (e.g., release height) are consistent with the other release categories in the MACCS2 modeling. The MACCS2 code is run to develop the 50 mile radial population dose for the intact containment core damage case.

Intact Containment Release Definition

MAAP case PNP_SLOCA_5 was developed and run in support of this RAI response. This MAAP case models a small loss of coolant accident (SLOCA), without high pressure safety injection (HPSI), but with containment heat removal available. Core damage and vessel breach occur in approximately 1.5 hours and 8.8 hours, respectively. The MAAP case was run out to 24 hours, at which time the total release fraction to the environment for Noble Gases and CsI were 8.2E-4 and 1.8E-5, respectively.

The MAAP plots for release to the environment were used to develop two plumes of ten hour duration each with release fractions as follow:

Release Fraction to Environment										
Plume	Xe/Kr	I	Cs	Te	Sr	Mo	La	Ce	Ba	U
#1	3.6E-4	1.7E-5	1.5E-5	1.9E-5	4.0E-8	3.1E-6	3.0E-9	1.0E-8	4.2E-7	3.0E-12
#2	4.6E-4	1.0E-6	1.0E-6	1.0E-6	7.0E-7	1.0E-7	6.6E-8	9.9E-7	3.2E-7	1.0E-8

Each plume is released at a height of 18.6 meters (consistent with the other Palisades release categories). Buoyant plume rise is neglected (zero plume heat) on the assumption that thermal heat will be quickly dissipated via the small assumed leakage pathways. The first plume is released at 1.5 hours, the same time as a General Emergency is declared (i.e., MACCS2 alarm time). The second plume is released at 11.5 hours. Release fractions for the second plume are based on MAAP values for a time of 24 hours.

Results

The 50 mile radial population dose calculated by MACCS2 for the Intact Containment case is 2.08E+3 person-Rem (2.08E+1 person-Sv).

This value is judged reasonable compared to the Late-Low Low population dose value of 4.10E+4 person-Rem. The difference between the Late-Low Low and Intact Containment population dose results is approximately a factor of 20. Population dose is significantly influenced by the Iodine group and Cesium group release fractions. The difference between the total release fractions of the Iodine group and the Cesium group between the Late-Low Low and Intact Containment release categories is approximately 11 and 25, respectively. These Iodine and Cesium release fraction relative factors of 11 and 25 are in the expected range to support the factor of 20 difference in the population dose (given that other fission product groups also influence the population dose, but to a lesser extent).

ATTACHMENT 2
RAI RESPONSE RELATED TO INFORMATION NEEDED CONCERNING CONTAINMENT TYPE A LEAK RATE TEST FREQUENCY

Request for additional information received by electronic mail September 2, 2011

Nuclear Regulatory Commission (NRC) Request

1. *In Section 4.3, "Supplemental Inspection Requirements," of the license amendment request (LAR), Table 4.3-1, "PLP Containment Inservice Inspection Periods (IWE/IWL)," only provides the typical inspection schedule that is based on Subsection IWE of the ASME Code, Section XI, and applies to the Class MC containment metallic liner and its attachments. Please provide a similar schedule of inspections, in a tabular format similar to that of Table 4.3-1, for a typical 15-year interval (between the last Type A test in refuel outage 1R15 and the proposed next Type A test in refuel outage 1R24) that were or will be performed on the containment concrete surfaces in accordance with Subsection IWL and/or other and explain how it meets the requirements in Section 9.2.3.2 of NEI 94-01, Revision 2-A (i.e., one inspection prior to each Type A test and at least three (3) other inspections between Type A tests for a total of four (4) inspections, if the Type A test interval is extended to 15 years), and Condition 2 in Section 4.1 of the NRC Safety Evaluation* [SE] *for topical report NEI 94-01, Revision 2-A.*

Entergy Nuclear Operations, Inc (ENO) Response

1. The requirements in Section 9.2.3.2 of NEI 94-01, Revision 2-A, and Condition 2 in Section 4.1, of the NRC SE for topical report NEI 94-01, Revision 2-A, have been met as described below. In accordance with the Palisades Nuclear Plant (PNP) Master In-Service Inspection Examination Plan, visual examinations of accessible concrete containment components, in accordance with ASME Code, Subsection IWL, are performed every five years, resulting in at least three IWL examinations being performed during a 15-year interval. As indicated in the LAR submittal, the proposed 15-year Type A test interval would be from May 3, 2001, to May 3, 2016. If approved, the next Type A test will be completed during the 1R24 refueling outage in 2015, approximately one year prior to the end of the 15-year interval. The next IWL examination, of the containment concrete, is planned in 2015, after the Type A examination has been completed. Two IWL examinations will have been completed prior to the next Type A test. These two IWL examinations were completed in 2005 and 2010.

   In addition to the IWL examinations, a visual inspection, of the accessible interior and exterior of the PNP containment building, was performed in 2009 and another inspection is scheduled to be performed prior to the Type A test in 2015, to satisfy the requirements of the 10 CFR 50 Appendix J Testing Program. These examinations are performed in sufficient detail to identify any evidence of deterioration, which may affect the structural integrity or leak tightness, of the containment building.

   Therefore, with the two IWL examinations performed in 2005 and 2010, the additional examination performed in 2009 (three examinations), and the general

visual examination scheduled to be performed in 2015 prior to the Type A test (fourth examination conducted prior to the next Type A test), the requirements in section 9.2.3.2 of NEI 94-01, Rev. 2-A would be met.

Condition 2 in Section 4.1 of the NRC SE for NEI 94-01, Rev 2-A, requires the submittal of a schedule of containment inspections to be performed prior to and between Type A tests. The following table provides the schedule for the containment surface visual examinations performed and planned from 2001 through 2015, assuming the Type A test frequency is extended to 15 years.

Calendar Year/Outage	Type A Test (ILRT)	Visual Examination of Accessible Exterior Surface/Type
2001/1R15	X	X / General Visual
2005		X / IWL Exam
2009		X / General Visual
2010		X / IWL Exam
2015/1R24	X	X / General Visual
2015		X / IWL Exam

NRC Request

2. *Condition 3 in Section 4.1 of the NRC Safety Evaluation (SE) for NEI 94-01, Revision 2-A, requires that licensees address the areas of the containment structure potentially subject to degradation (Refer to SE Section 3.1.3). Section 3.1.3 of the NRC SE, in part, states that licensees referencing NEI 94-01, Revision 2, in support of a request to amend their TS should also explore/consider such inaccessible degradation-susceptible areas in plant-specific inspections, using viable, commercially available NDE methods (such as, boroscopes, guided wave techniques, etc.– see Report ORNL/NRC/LTR-02/02, "Inspection of Inaccessible Regions of Nuclear Power Plant Containment Metallic Pressure Boundaries," June 2002 (ADAMS Accession No.ML061230425) for recommendations) to support plant-specific evaluations.*

   The NRC staff's intent of this statement in the SER is that licensees should explore and consider NDE techniques, such as those discussed in the reference or other state-of-the-art methods for inspections of inaccessible degradation-susceptible areas of the containment pressure boundary, in a proactive manner to support plant-specific evaluations of inaccessible areas, as these advanced technologies become commercially available and viable for implementation in

practice in the future. The issue related to inaccessible areas is especially important in light of several instances of significant through-wall containment liner degradations (corrosion) that have been identified in the last decade in US operating nuclear plants, where the corrosion initiated at the inaccessible concrete-steel interface. While recognizing that these techniques may not be fully commercially viable at the present time, in order to fully address Condition 3 in Section 4.1 of the NRC SE, the licensee is requested to provide the following plant-specific information:

a) Identify the specific areas of the PNP containment pressure boundary (both concrete and steel) that are inaccessible and susceptible to degradation.

ENO Response

2.a) In accordance with the PNP IWE program, the inaccessible portions of the containment building liner plate are as follows:

- The portions of the liner plate below the concrete floor at the 590-foot elevation,
- Leak chase channels used to check the seams in the floor liner plate, which are embedded on the inaccessible side of the containment liner plate,
- Areas of the containment sump penetrations to containment liner connections which are embedded in concrete, and
- The portion of the fuel transfer tube which is encased in concrete or buried in the "gravel pit."

From the PNP IWL program, the inaccessible areas are as follows:

- The portions of the concrete surface that are covered by the liner, foundation material, and backfill or are otherwise obstructed by adjacent structures, components, parts or appurtenances.

NRC Request

2.b) Acknowledge that NDE technologies (such as those described in the reference or others) would be explored in a proactive manner and considered in the future for the examination of inaccessible degradation-susceptible areas of the containment, as these technologies become commercially viable for implementation in practice. Include information of activities (such as participation in applicable cognizant owner's groups, code committees, professional societies, etc. in the field) that the licensee is proactively participating in or plans to participate in for tracking applicable ongoing technology developments, operating experience and contributing towards achieving this goal.

ENO Response

2.b) As noted in 2.a above, there are areas that the PNP IWE/IWL programs have identified as inaccessible. As discussed in 2.c below, PNP had an examination that identified a condition, in an accessible area above the moisture barrier, around the 590-foot elevation of the containment that suggested the potential for degradation in an inaccessible area. Subsequent examination below the moisture barrier found no unsatisfactory degradation.

ENO actively participates in various nuclear utility owner's groups, ASME Code committees, and with NEI, to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its applicability to PNP. New commercially available technologies, for the examination of the inaccessible areas of containment, are explored and considered as part of these activities.

NRC Request

2.c) Provide information of instances, if any, during implementation of the containment ISI program at PNP in accordance with IWE/IWL where existence of or potential for degraded conditions in inaccessible areas of the concrete containment structure and metallic liner were identified and evaluated based on conditions found in accessible areas, as required by 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A). If there were any instances of such conditions, please discuss the findings and actions taken.

ENO Response

2.c) Prior to the implementation of the PNP IWE/IWL programs, an indication of general corrosion was discovered, during the 1998 refueling outage, 1R13, above the moisture barrier around the containment circumference at the 590-foot elevation inside containment. This condition was documented and dispositioned. The corrective actions required investigation below the moisture barrier in the location containing indications as well as other locations. Boroscope examinations determined that no unsatisfactory degradation existed, in the inaccessible area beneath the moisture barrier. To date, there have been no additional instances in relation to the IWE or the IWL examinations where conditions were identified in accessible areas that would indicate the potential presence of degradation in the inaccessible areas. The PNP IWE/IWL programs contain requirements to evaluate the acceptability of the inaccessible areas, if such conditions were identified, in accordance with 10 CFR 50.55a(b)(2)(ix)(A) and 10 CFR 50.55a(b)(2)(viii)(E).

NRC Request

3. *In order to adequately address Condition 4 in Section 4.1 of the NRC Safety Evaluation for Topical Report NEI 94-01, Revision 2-A, the licensee is requested to provide information that will demonstrate its understanding of: (a) the post-modification pressure testing and examination required to be performed prior to return to service following "major" containment modifications; and (b) the distinction between major and minor containment modifications. The NRC staff position is that the pressure testing performed prior to return to service following any "major" containment modification should demonstrate both structural integrity and leak-tight integrity of the containment. In responding to this RAI, please refer to Section 3.1.4 of the NRC Safety Evaluation for Topical Report NEI 94-01, Revision 2-A, and the regulatory condition in paragraph 10 CFR 50.55a(b)(2)(ix)(J) and its statement of considerations in the final rule published on June 21, 2011 in the Federal Register, Vol. 76, No. 119, pp 36232 thru 36279 (76 FR 36232-36270).*

ENO Response

3. As indicated in the Federal Register, on June 21, 2011, the NRC amended paragraph 10CFR50.55a(b)2(ix) to add a new paragraph (b)(2)(ix)(J) to address pressure testing requirements following major modifications of Class MC containment structures when applying Article IWE-5000, of Subsection IWE of the 2007 Edition to the latest edition and addenda incorporated by reference into 50.55a (currently the 2008 Addenda of the ASME B&PV Code, Section XI). PNP is currently in the second ten-year interval for the IWE/IWL Program. As indicated in the LAR submittal Section 4.3 (page 13 of 27), the applicable code edition and addenda for the second ten-year interval IWE/IWL program is the 2004 Edition. There are no relief requests associated with this interval.

   As indicated in Section 4.1 (page 9 of 27), of the LAR, replacement of the steam generators occurred during the 1990 – 1991 refueling outage. A construction opening was cut through the containment wall to allow replacing the steam generators. The construction opening was considered a major repair to the containment building. A Type A integrated leak rate test (ILRT) and a structural integrity test (SIT), rather than a periodic containment Type A ILRT, were completed, to show that the repairs to the containment adequately met the Technical Specifications leakage requirements. The Type A ILRT and SIT are documented in the PNP Final Safety Analysis Report, Sections 5.8.9.6.1 and 5.8.9.6.2. The results of the SIT demonstrated that the containment was fully restored to the design condition existing prior to the steam generator replacement.

As indicated in Section 4.0 (page 7 of 27) of the April 6, 2011, LAR submittal, there are no planned modifications that would require a Type A test prior to the 2015 refueling outage, 1R24, when the next Type A test would be performed under the proposed change. Any unplanned modifications to the containment prior to the next scheduled Type A test would be subject to the special testing requirements of Section IV.A of 10 CFR 50, Appendix J. It is understood that a repair or modification of the containment would require classification of the repair or modification using the guidance in Section 3.1.4 of the NRC Safety Evaluation for Topical Report NEI 94-01, Revision 2-A. In addition, it is understood that the NRC staff has agreed to a relief request from the IWE requirements for performing the Type A test and has accepted a combination of actions as listed in Section 3.1.4 of the NRC Safety Evaluation for Topical Report NEI 94-01, Revision 2-A, defined as a short duration structural test, which may be performed in place of the Type A test. For minor modifications, leakage integrity of the affected pressure retaining areas should be verified by a local leak rate test.

NRC Request

4. *Please address whether bellows used on penetrations through containment pressure-retaining boundaries at PNP. If so, since degradation of bellows is a source for potential leakage, please provide information on their type (single-ply or two-ply), location, inspection, testing and operating experience with regard to detection of leakage through penetration bellows.*

ENO Response

4. As indicated in the LAR submittal, Section 4.2 (page 11 of 27), the PNP piping and ventilation penetrations are the rigid welded type and are solidly anchored to the containment wall, thus precluding any requirement for expansion bellows.