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Fax: 440-280-8029August 10, 2012
L-12-293

10 CFR 50.55a

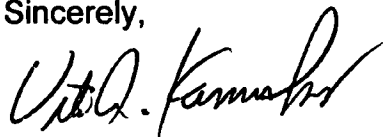
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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
Response to Request for Additional Information Related to Request for Alternative
Examination Requirements for American Society of Mechanical Engineers (ASME)
Class 1 Piping Welds (TAC No. ME7564)

By letter dated November 14, 2011 (Accession No. ML113180450), FirstEnergy Nuclear Operating Company (FENOC) submitted a request for Nuclear Regulatory Commission (NRC) approval for continued use of the existing Perry Nuclear Power Plant (PNPP) risk-informed inservice inspection (RI-ISI) program, with updates, relative to certain non-destructive examination requirements associated with ASME Class 1 piping welds. By letter dated July 16, 2012 (Accession No. ML12167A313), the NRC staff requested additional information to complete its review. Responses to the NRC staff's questions are provided in the attachment.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Phil H. Lashley, Supervisor – Fleet Licensing, at (330) 315-6808.

Sincerely,



Vito A. Kaminskas

Attachment: Response to July 16, 2012 Request for Additional Information

cc: NRC Region III Administrator
NRC Resident Inspector
Nuclear Reactor Regulation Project Manager

Attachment
L-12-293

Response to July 16, 2012 Request for Additional Information
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By letter dated November 14, 2011, FirstEnergy Nuclear Operating Company (FENOC) submitted a 10 CFR 50.55a(a)(3) request for Nuclear Regulatory Commission (NRC) review and approval. By letter dated July 16, 2012, the NRC staff requested additional information to complete its review. The NRC staff's questions are presented below in bold type, followed by FENOC's responses.

REQUEST FOR ADDITIONAL INFORMATION

1. **The submittal indicates that a full-scope peer review was performed in 1997 for the internal events probabilistic risk assessment (PRA) and gap or self-assessments have been performed periodically since the 1997 peer review. The submittal also indicates that the PRA model has been revised periodically and some of these revisions included changes to address findings from the 1997-related review and self-assessments.**
 - a. **Regarding any changes since the independent full-scope peer review characterized as a PRA upgrade per American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS)-RA-Sa-2009, identify if a focused-scope peer review was performed for these changes consistent with the guidance in ASME/ANS-RA-Sa-2009, as endorsed by NRC Regulatory Guide 1.200, and describe any findings from that focused-scope peer review and the resolution of these findings for this application.**
 - b. **If a focused-scope peer review has not been performed for changes characterized as a PRA upgrade, describe what actions will be implemented to address this review deficiency. State when the application will be supplemented to describe any findings from that focused-scope peer review and the resolution of these findings for this application.**

Response:

An independent peer review of the PNPP PRA model was performed in 1997 under the auspices of the Boiling Water Reactor Owner's Group probabilistic safety analysis (PSA) peer review certification process. Multiple internal self-assessments, including a 2008 self-assessment utilizing an independent contractor, have also been performed. During the 2008 review, ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," December 2005, was used as the standard for the

review. The purpose of this review was to determine gaps between the model at the time and the standard's requirements.

Following the February 2011 update of the PNPP average-maintenance model that included the incorporation of review (gap) findings from the 2008 review, the model is judged to meet Capability Category II for all supporting requirements regarding Level 1 internal events only, and is also judged to be compliant with Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, January 2007, with the exception of internal flooding and large early release frequency (LERF) modeling. Additionally, the latest model update was structured to satisfy the supporting requirements of the PRA Standard, ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009, for Level 1 internal events, minus internal flooding and LERF.

At the time of the 10 CFR 50.55a(a)(3) submittal, the internal flooding and Level 2 LERF models were judged not to be of the quality required to satisfy the requirements expected per Regulatory Guide 1.200, and subsequently were not utilized for that purpose. Efforts have been expended since the submittal to upgrade the quality of these two models. The Level 2 LERF model has undergone a Regulatory Guide 1.200 (focused scope) peer review in mid-2011; however, that model has not been made effective. Internal flooding upgrade efforts are currently ongoing, with a focused scope peer review completed in July 2012 and gap closure in progress. Following peer review comment incorporation, both the internal flooding and Level 2 LERF models are to become effective.

The model updates performed on the Level 1 model since the PSA peer review certification process have not utilized new methodologies; therefore, a PRA upgrade status, as defined by the PRA standard, would not apply, and as such, a full scope peer review was not performed.

- 2. The submittal indicates that the licensee's PRA model does not include large early release frequency (LERF). Although a bounding analysis may meet the criteria set forth in the standard, this does not account for high LERF values that may exist and could have an impact on conditional large early release probability (CLERP). Provide the analysis of CLERP based on a realistic approach to LERF calculation.**

Response:

As discussed above, the Level 2 LERF models were judged not to be of the quality required to satisfy the requirements of Regulatory Guide 1.200 standards at the time of the submittal (November 14, 2011). As such, a bounding LERF analysis was subsequently performed that showed that the risk acceptance criteria continued to be satisfied. Table 1-A depicts the differences between the last deterministic Section XI

program and the submittal with corresponding CDF and LERF impacts. As demonstrated in Table 1-A, with the bounding case of CLERP set equal to 1 (that is, all core damage events lead to a large early release), LERF is equal to the CDF.

3. **In a prior risk-informed inservice relief request submitted on February 12, 2001, the licensee used a Level 2 PRA to perform LERF calculations and produced a table highlighting the change in risk for both core damage frequency and LERF when compared to the last deterministic ASME Section XI program. Provide delta risk calculations for both core damage frequency and LERF on a system and total basis when compared to the last deterministic ASME, Section XI, program for NRC staff consideration.**

Response:

Table 1-A depicts the differences between the last deterministic ASME Section XI program and this risk-informed inservice inspection (RI-ISI) submittal with corresponding CDF and LERF impacts.

Table 1-A

System	Risk ¹		Consequence Rank	Failure Potential ¹		Inspection Locations ²			CDF Impact		LERF Impact ³	
	Category	Rank		Damage Mechanisms	Rank	Section XI	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
1B13	2 (1)	High (High)	High	TASCS, TT (IGSCC, FAC)	Medium (High)	6	1	-5	2.41E-10	6.70E-10	2.41E-10	6.70E-10
1B13	2	High	High	TT	Medium	0	1	1	-2.41E-10	-1.34E-10	-2.41E-10	-1.34E-10
1B13	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	17	3	-14	9.38E-11	9.38E-11	9.38E-11	9.38E-11
1B13	4	Medium	High	None	Low	9	0	-9	6.03E-11	6.03E-11	6.03E-11	6.03E-11
1B13	6a (5a)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	4	0	-4	2.00E-12	2.00E-12	2.00E-12	2.00E-12
1B13	6a	Low	Medium	None	Low	1	0	-1	5.00E-13	5.00E-13	5.00E-13	5.00E-13
1B13 Total									1.57E-10	6.93E-10	1.57E-10	6.93E-10
1B21	4	Medium	High	None	Low	22	8	-14	9.38E-11	9.38E-11	9.38E-11	9.38E-11
1B21	6a	Low	Medium	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1B21	7a	Low	Low	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1B21 Total									9.38E-11	9.38E-11	9.38E-11	9.38E-11
1B33	4	Medium	High	None	Low	28	12	-16	1.07E-10	1.07E-10	1.07E-10	1.07E-10
1B33 Total									1.07E-10	1.07E-10	1.07E-10	1.07E-10
1C41	4	Medium	High	None	Low	0	1	1	-6.70E-12	-6.70E-12	-6.70E-12	-6.70E-12
1C41	6a	Low	Medium	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1C41 Total									-6.70E-12	-6.70E-12	-6.70E-12	-6.70E-12
1E12	2	High	High	TT	Medium	2	1	-1	-8.04E-11	1.34E-10	-8.04E-11	1.34E-10
1E12	4	Medium	High	None	Low	18	9	-9	6.03E-11	6.03E-11	6.03E-11	6.03E-11
1E12	6a	Low	Medium	None	Low	10	0	-10	5.00E-12	5.00E-12	5.00E-12	5.00E-12
1E12	6b	Low	Low	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1E12 Total									-1.51E-11	1.99E-10	-1.51E-11	1.99E-10
1E21	4	Medium	High	None	Low	7	3	-4	2.68E-11	2.68E-11	2.68E-11	2.68E-11
1E21	6a	Low	Medium	None	Low	2	0	-2	1.00E-12	1.00E-12	1.00E-12	1.00E-12
1E21 Total									2.78E-11	2.78E-11	2.78E-11	2.78E-11

System	Risk ¹		Consequence Rank	Failure Potential ¹		Inspection Locations ²			CDF Impact		LERF Impact ³	
	Category	Rank		Damage Mechanisms	Rank	Section XI	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
1E22	4	Medium	High	None	Low	8	3	-5	3.35E-11	3.35E-11	3.35E-11	3.35E-11
1E22	6a	Low	Medium	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1E22 Total									3.35E-11	3.35E-11	3.35E-11	3.35E-11
1E32	4	Medium	High	None	Low	0	2	2	-1.34E-11	-1.34E-11	-1.34E-11	-1.34E-11
1E32 Total									-1.34E-11	-1.34E-11	-1.34E-11	-1.34E-11
1E51	2	High	High	TT	Medium	2	2	0	-3.22E-10	0.00E+00	-3.22E-10	0.00E+00
1E51	4	Medium	High	None	Low	2	1	-1	6.70E-12	6.70E-12	6.70E-12	6.70E-12
1E51	5a	Medium	Medium	TT	Medium	7	3	-4	-1.20E-11	4.00E-11	-1.20E-11	4.00E-11
1E51	6a	Low	Medium	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1E51 Total									-3.27E-10	4.67E-11	-3.27E-10	4.67E-11
1G33	4	Medium	High	None	Low	7	2	-5	3.35E-11	3.35E-11	3.35E-11	3.35E-11
1G33	6a	Low	Medium	None	Low	11	0	-11	5.50E-12	5.50E-12	5.50E-12	5.50E-12
1G33	7a	Low	Low	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1G33 Total									3.90E-11	3.90E-11	3.90E-11	3.90E-11
1N22	2	High	High	TT	Medium	0	1	1	-2.41E-10	-1.34E-10	-2.41E-10	-1.34E-10
1N22	5a	Medium	Medium	TT	Medium	0	7	7	-1.26E-10	-7.00E-11	-1.26E-10	-7.00E-11
1N22	6b	Low	Low	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1N22 Total									-3.67E-10	-2.04E-10	-3.67E-10	-2.04E-10
1N27	2 (1)	High (High)	High	TASCS, TT (FAC)	Medium (High)	10	5	-5	-4.02E-10	6.70E-10	-4.02E-10	6.70E-10
1N27	2 (1)	High (High)	High	TT (FAC)	Medium (High)	7	10	3	-1.85E-09	-4.02E-10	-1.85E-09	-4.02E-10
1N27	4 (1)	Medium (High)	High	None (FAC)	Low (High)	0	1	1	-6.70E-12	-6.70E-12	-6.70E-12	-6.70E-12
1N27	5a (3)	Medium (High)	Medium	TT (FAC)	Medium (High)	0	1	1	-1.80E-11	-1.00E-11	-1.80E-11	-1.00E-11
1N27 Total									-2.28E-09	2.51E-10	-2.28E-09	2.51E-10
Grand Total									-2.55E-09	1.27E-09	-2.55E-09	1.27E-09

Notes:

1. Risk categorization is presented with and without consideration of FAC and IGSCC damage mechanisms; parentheses enclose those with consideration of FAC and IGSCC.
2. As indicated in Table 1 of the November 14, 2011 submittal, the inspection location population may include welds selected from both Code Categories B-F and B-J.
3. In the bounding case of CLERP set equal to 1, LERF equals CDF.

Table 1-A Terms:

Systems

1B13 – Reactor Pressure Vessel
Nozzles and Connections
1B21 – Nuclear Boiler – Main Steam

1B33 – Reactor Recirculation
1C41 – Standby Liquid Control
1E12 – Residual Heat Removal
1E21 – Low Pressure Core Spray

CDF Impact

w/ POD – with Probability of Detection
w/o POD – without Probability of Detection

1E22 – High Pressure Core Spray

1E32 – Main Steam Isolation Valve Leakage Control
System
1E51 – Reactor Core Isolation Cooling
1G33 – Reactor Water Cleanup
1N22 – Miscellaneous Drains – Main Steam Drains
1N27 – Feedwater

LERF Impact

w/ POD – with Probability of Detection
w/o POD – without Probability of Detection

Degradation Mechanisms

FAC – Flow Accelerated Corrosion

IGSCC – Intergranular Stress Corrosion Cracking

TASCS – Thermal Stratification, Cycling and Striping
TT – Thermal Transient