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DTE Energy
Detroit Edison

10 CFR 52.79
10 CFR 2.390
10 CFR 9.17

July 29, 2011
NRC3-11-0028

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

References:
1) Fermi 3
Docket No. 52-033
2) Letter from Jerry Hale (USNRC) to Jack M. Davis (Detroit Edison), "Request for Additional Information Letter No. 53 Related to the SRP Section 10.02.03 for the Fermi 3 Combined License Application," dated March 28, 2011
3) Letter from Peter W. Smith (Detroit Edison) to USNRC, "Detroit Edison Company Response to NRC Request for Additional Information Letter No. 53," NRC3-11-0012, dated April 27, 2011

Subject: Detroit Edison Company Supplemental Response to NRC Request for Additional Information Letter No. 53

In Reference 2, the NRC requested additional information to support the review of certain portions of the Fermi 3 Combined License Application (COLA). The responses to the Requests for Additional Information (RAIs) in Reference 2 related to turbine rotor integrity were provided in Reference 3.

NRC staff has requested supplemental information regarding RAIs 10.02.03-12 through 10.02.03-16. Supplemental information within these responses is indicated by shaded text. The responses to these RAIs contain GE proprietary information, and as such, both proprietary and non-proprietary versions of the responses are provided. Non-proprietary responses are provided in Enclosures 1 through 5 of Attachment 1. The proprietary versions of the responses are provided in Enclosures 1 through 5 of Attachment 6. Proprietary information within these responses is indicated by double brackets.

Additionally, this letter provides Revision 3 of the of GE-Energy Steam Turbines (GE-ST) report ST-56834, "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis." The non-proprietary version of the report, ST-56834/N-P, is provided in Attachment 2.



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The proprietary version of the report, ST-56834/P, is provided in Attachment 7. COLA Markups that incorporate the updated revision of the report are provided in Attachment 3.

As noted above, Attachments 6 and 7 contain GE proprietary information as defined by 10 CFR 2.390. Affidavits are included in Attachments 4 and 5 that identify the information contained in Attachments 6 and 7, respectively, as proprietary to GE. Detroit Edison and GE request that Attachments 6 and 7 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 10 CFR 9.17.

If you have any questions, or need additional information, please contact me at (313) 235-3341.

I state under penalty of perjury that the foregoing is true and correct. Executed on the 29th day of July 2011.

Sincerely,

Peter W. Smith, Director
Nuclear Development – Licensing and Engineering
Detroit Edison Company

Attachments:
1) Supplemental Response to RAI Letter No. 53 (Questions 10.02.03-12 through -16) [Public Version]
2) GE-ST "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis," ST-56834/N-P, Revision 3 [Public Version]
3) COLA Markups [Public]
4) Affidavit of Damodar Padhi (GE) for RAI Responses, dated July 25, 2011 [Public]
5) Affidavit of Damodar Padhi (GE) for ST-56834/P, dated July 25, 2011 [Public]
6) Supplemental Response to RAI Letter No. 53 (Questions 10.02.03-12 through -16) [Non-Public Version]
7) GE-ST "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis," ST-56834/P, Revision 3 [Non-Public Version]

cc: (see next page)

cc: Adrian Muniz, NRC Fermi 3 Project Manager
Michael Eudy, NRC Fermi 3 Project Manager
Raj Anand, NRC Fermi 3 Project Manager (w/o attachments)
Jerry Hale, NRC Fermi 3 Project Manager (w/o attachments)
Bruce Olson, NRC Fermi 3 Environmental Project Manager (w/o attachments)
Fermi 2 Resident Inspector (w/o attachments)
NRC Region III Regional Administrator (w/o attachments)
NRC Region II Regional Administrator (w/o attachments)
Supervisor, Electric Operators, Michigan Public Service Commission (w/o attachments)
Michigan Department of Natural Resources and Environment
Radiological Protection Section (w/o attachments)

**Attachment 1**
**NRC3-11-0028**

**Supplemental Response to RAI Letter No. 53**
**(Questions No. 10.02.03-12 through -16)**
**[Public Version]**

(following 16 pages)

**Attachment 1**
**NRC3-11-0028**

**Enclosure 1**

**Response to RAI Letter No. 53**
**(eRAI Tracking No. 5608)**

**RAI Question No. 10.02.03-12**

**NRC RAI 10.02.03-12**

*In a letter dated October 5, 2010, the applicant's response did not fully provide the information requested in RAI 4641, Question 10.02.03-4. Therefore, as requested in RAI 4641, Question 10.02.03-4, provide the operational experience of this turbine material as an integral rotor, which should include how many rotors, units, operating hours, number of defects detected during inspections, rotor failures, etc. to provide a basis of the material to be used. Also, the RAI response stated that "specification B50A373B8 or an equivalent specification with more restrictive chemistry requirements" will be used for the LP rotors. Does the more restrictive chemistry only apply to elements that have deleterious effects on toughness, such as sulfur and phosphorus as stated in NUREG-0800, Section 10.2.3, or does it also apply to the alloying elements? If it applies to the alloying elements, then this equivalent specification should be submitted to the staff for review as outlined in NUREG-0800, Section 10.2.3, Paragraph III.1.*

**Supplemental Response**

Since the 1980s, all General Electric solid (i.e., not shrunk-on wheel) nuclear low-pressure (LP) rotors have been manufactured in accordance with GE specification B50A373B8. To date, [[ ]] rotor-years. No rotor failures have occurred within this fleet of units.

GE's nuclear LP monoblock experience can be divided into three design generations according to the table below.

[[ ]]

The rotor forging material and chemistry remains unchanged throughout the generations of rotors. What is changed is the geometry of the bucket to wheel attachment to reduce stresses and the use of metal improvement processes (shot-peening).

Early monoblock rotors (Generation 1) were designed to be direct replacements for the built-up rotors that were originally supplied with the turbine. As such, no changes to the bucket and wheel attachment (dovetail) geometry were made. Some stress corrosion cracking (SCC) has been found in the Generation 1 fleet.

GE redesigned the dovetail geometry in the early 1990s to reduce the stresses and added shot-peening as a standard process. The changes to the dovetail geometry were limited by the requirement for re-use of existing buckets. Inspection results of the Generation 2 solid rotors indicates that the shot-peening and the geometry change has eliminated or at least significantly delayed the initiation of SCC.

The current design (Generation 3) monoblock rotors include significant geometric changes to further reduce peak tensile stresses. Shot-peening continues to be standard practice. The first Generation 3 monoblocks are yet to be inspected.

General Electric has changed the material specification for nuclear LP monoblock rotors from B50A373B8 to B50A373B12. The new material specification places tighter control on nickel content. There are no changes to the elements that have deleterious effects on toughness, such as sulfur and phosphorus as stated in NUREG-0800, Section 10.2.3.

The B50A373B8 specification allows a range of [[ ]] nickel, which covers both small fossil LP applications and nuclear LP applications. The B50A373B12 specification allows a range of [[ ]] nickel. All nuclear monoblock rotor forgings manufactured to date were manufactured in the nickel range of [[ ]]. As the rotor forging supply base further develops (i.e., additional monoblock forging capacity coming online), it is prudent that GE specifies chemistry requirements which are reflective of the nickel range required to achieve properties in the nuclear monoblock forgings.

The B50A373 material specification has been revised to include B50A373B12. The specification will be available for review.

**Proposed COLA Revision**

None

**Attachment 1**
**NRC3-11-0028**

**Enclosure 2**

**Response to RAI Letter No. 53**
**(eRAI Tracking No. 5608)**

**RAI Question No. 10.02.03-13**

**NRC RAI 10.02.03-13**

*In a letter dated October 5, 2010, the applicant's response to RAI 4641, Question 10.02.03-6 stated that a historical Fracture Appearance Transition Temperature (FATT) value was used in the turbine missile analysis. However, ESBWR DCD, Section 10.2.3.1.2 states that the material for the rotors will have a maximum 50% FATT value of +30°F. Therefore, the bounding turbine missile probability analysis (GE-Energy Steam Turbines (GE-ST) report ST-56834/P) should be based on the bounding material properties of the ESBWR DCD (50% FATT value of +30°F) in lieu of historical FATT measurements (50% FATT value of -30°F) currently used in the GE-Energy Steam Turbines (GE-ST) report ST-56834/P. Furthermore, Sections 10.2.3.8 and 10.2.5 of the ESBWR DCD states that the COL applicant will provide the turbine missile probability analysis, and if the actual material properties of the as-built turbine are not available, the bounding material property values should be used. Therefore, since the as-built turbine rotor material properties for Fermi, Unit 3 are not known, GE-Energy Steam Turbines (GE-ST) report ST-56834/P should use the bounding material properties of the ESBWR DCD.*

**Supplemental Response**

Missile generation probability analysis has been updated to include the requested bounding assumption of +30°F FATT. More specifically, the previous assumption of normally distributed FATT was eliminated and replaced by the assumption of a fixed (non-variant) FATT of +30°F. Figures 8-1 and 9-1 from GE-ST report ST-56834/P were updated to reflect the impact of the requested bounding FATT assumption and are included below.

Referring to the updated figures, the flat regions of both remain associated with tensile failure during gross abnormal over speed of the turbine and, as such, are not impacted by the change in assumed FATT.

The change in assumed FATT does, however, affect the turn-up region of each curve. The turn-up region is associated with the probability of a dovetail SCC crack reaching critical size. Regardless of FATT assumption, a certain amount of time is required before the burst mode associated with an SCC crack reaching critical flaw size is more likely vs. the tensile/over speed burst failure mode. The steady increase in probability after the transition in limiting burst mode is associated with time dependent SCC crack growth.

As shown in the figures, the impact of the requested +30°F bounding FATT assumption is critical flaw size being reached in just over 40 years. The predicted average SCC crack depth of [[ ]] years of accumulated turbine operation is within the crack detection capability for the planned testing described in GE-ST report ST-56834/P, Section 10. Therefore, GE will update the probability analysis to reflect the as-shipped rotor FATT and then after each in-service inspection to reflect the both the actual measured SCC crack depth (if a crack is found) and the expected uncertainty in the associated measurement. Results from the updated analysis will dictate future re-inspection frequency necessary to ensure that NRC limits on missile probability are not exceeded.

Figure 8-1, Unit Featuring Solid Rotors: Annual Missile Probability

[[ ]]

Figure 9-1, Unit Featuring Bored Rotors: Annual Missile Probability

[[ ]]

**Proposed COLA Revision**

None

**Attachment 1**
**NRC3-11-0028**

**Enclosure 3**

**Response to RAI Letter No. 53**
**(eRAI Tracking No. 5608)**

**RAI Question No. 10.02.03-14**

**NRC RAI 10.02.03-14**

*In a letter dated October 5, 2010, the applicant's response did not fully provide the information requested in RAI 4641, Question 10.02.03-8, and therefore the following information is requested:*

*a. Part (b) of the response to RAI 4641, Question 10.02.03-8 does not provide the quantitative information requested about flaw size and detection capability. Rather, it states that volumetric inservice inspection of solid LP rotors is unnecessary. Therefore, as requested in RAI 4641, Question 10.02.03-8, discuss the operating experience of solid rotors, including the effects on material properties and whether current volumetric inspections can detect cracking before they reach critical size resulting in a turbine missile. Compare the flaw size capability of the volumetric inspections to the average undetected embedded flaw specified in Section 4.2.2.*

*b. Section 10.2.3.6 of the ESBWR DCD states that volumetric inservice inspection of the rotor will be performed. However, the response to part (b) of the response to RAI 4641, Question 10.02.03-8, states "inservice volumetric inspection of solid nuclear LP rotors is not required to meet the calculations included in the report [GE-Energy Steam Turbines (GE-ST) report ST-56834/P]". Provide an analysis and discussion for a surface flaw that could grow radially inward and cause a rupture of the LP rotor in the locations (other than in the dovetail regions) where an inservice volumetric inspection is not performed. Otherwise, a volumetric inspection of the LP rotor should be included in the turbine inservice inspection program as outlined in NUREG-0800, Section 10.2.3, Paragraph II.5.*

**Supplemental Response**

> *a. Part (b) of the response to RAI 4641, Question 10.02.03-8 does not provide the quantitative information requested about flaw size and detection capability. Rather, it states that volumetric inservice inspection of solid LP rotors is unnecessary. Therefore, as requested in RAI 4641, Question 10.02.03-8, discuss the operating experience of solid rotors, including the effects on material properties and whether current volumetric inspections can detect cracking before they reach critical size resulting in a turbine missile. Compare the flaw size capability of the volumetric inspections to the average undetected embedded flaw specified in Section 4.2.2.*

As discussed in the response to RAI 10.02.03-12, operational issues with GE solid rotors have been limited to dovetail SCC in early Generation I designs.

GE-ST report ST-56834/P includes consideration of center core material properties. Center cores removed from monoblock rotors are tested extensively. These test results are the statistical basis for the deep-seated material properties assumed in the report.

The critical flaw size of GE monoblock rotors is quite large. Outside surface geometry and features, however, limit the extent to which solid rotors can be inspected during an in-service volumetric test. At locations where sufficient access exists, an external volumetric inspection process can detect cracking before critical flaw size is reached. External surface features, however, limit the extent of inspectability.

Since the external geometry of a steam turbine rotor does not permit 100% volumetric in-service inspection, the GE process places tight controls on the rotor metallurgy and pre-service inspection.

As discussed in Sections 8 and 9 of GE-ST report ST-56834/P, the annual probability of missile generation is dominated by turbine over speed for the first 20 years of life, then postulated SCC crack growth originating at the axial entry dovetail slot bottoms thereafter. The annual probability of generating a missile from an undetected flaw growing to critical crack size is never the most limiting factor and is always much less than the NRC annual probability for the entire 60-year life.

*b. Section 10.2.3.6 of the ESBWR DCD states that volumetric inservice inspection of the rotor will be performed. However, the response to part (b) of the response to RAI 4641, Question 10.02.03-8, states "inservice volumetric inspection of solid nuclear LP rotors is not required to meet the calculations included in the report [GE-Energy Steam Turbines (GE-ST) report ST-56834/P]". Provide an analysis and discussion for a surface flaw that could grow radially inward and cause a rupture of the LP rotor in the locations (other than in the dovetail regions) where an inservice volumetric inspection is not performed. Otherwise, a volumetric inspection of the LP rotor should be included in the turbine inservice inspection program as outlined in NUREG-0800, Section 10.2.3, Paragraph II.5.*

GE-ST report ST-56834/P as-submitted includes analysis and discussion of a worst-case surface flaw that could grow radially inward and cause a rupture of the LP rotor in locations other than the dovetail region. The bored rotor surface stress shown in Table 4-1 (stage 1) is the maximum predicted surface stress for the entire LP rotor. The total predicted stage 1 tangential stress magnitude of [[ ]] (found by adding the values shown in the 2nd and 3rd columns) exceeds the magnitude predicted along the entire outer surface including the axial entry dovetail slot bottoms. The overall missile probability summarized in Figure 9-1 includes the probability of an escaping bore surface flaw at this peak surface stress location (reference Section 4.2.2) reaching critical size and resulting in an uncontained missile.

**Proposed COLA Revision**

None

**Attachment 1**
**NRC3-11-0028**

**Enclosure 4**

**Response to RAI Letter No. 53**
**(eRAI Tracking No. 5608)**

**RAI Question No. 10.02.03-15**

**NRC RAI 10.02.03-15**

*In a letter dated October 5, 2010, the applicant's response did not fully provide the information requested in RAI 4641, Question 10.02.03-10, and therefore the following information is requested:*

*a. As requested in RAI 4641, Question 10.02.03-10, provide the tangential stresses at the slot bottoms of axial entry dovetails in Section 4.3 of the GE-Energy Steam Turbines (GE-ST) report ST-56834/P and compare them to the corresponding stresses around the previous shrunk-on-wheel keyways for a similar size turbine to demonstrate that the ESBWR axial entry dovetail slot bottoms feature dramatically lower tangential stresses versus shrunk-on-wheel keyways, and therefore the use of shrunk-on-wheel crack initiation and growth characteristics is considered conservative.*

*b. Concerning the location of axial entry dovetails, clarify which stages are axial entry dovetails since Sections 10.1.1 and 4.3 of the GE-Energy Steam Turbines (GE-ST) report ST-56834/P identifies different stages that are axial entry dovetails.*

*c. Provide operating experience with shot-peening of a rotor which demonstrates that compressive stresses are created and increases initiation time for this material and geometry.*

**Supplemental Response**

*a. As requested in RAI 4641, Question 10.02.03-10, provide the tangential stresses at the slot bottoms of axial entry dovetails in Section 4.3 of the GE-Energy Steam Turbines (GE-ST) report ST-56834/P and compare them to the corresponding stresses around the previous shrunk-on-wheel keyways for a similar size turbine to demonstrate that the ESBWR axial entry dovetail slot bottoms feature dramatically lower tangential stresses versus shrunk-on-wheel keyways, and therefore the use of shrunk-on-wheel crack initiation and growth characteristics is considered conservative.*

[[ ]]

*b. Concerning the location of axial entry dovetails, clarify which stages are axial entry dovetails since Sections 10.1.1 and 4.3 of the GE-Energy Steam Turbines (GE-ST) report ST-56834/P identifies different stages that are axial entry dovetails.*

As shown in Figure 4-1 of GE-ST report ST-56834/P, stages 5, 6, and 7 are axial entry dovetail designs. By comparison, stages 1-4 feature tangential entry dovetails. There is a typographical error in Section 10.1.1, "Rotor Dovetail Inspections," of report ST-56834/P, Revision 2. The text should read: "Surface inspection of tangential entry dovetails (stages 1 thru 4)," not stages 1 thru 5. This error has been corrected in ST-56834/P, Revision 3.

*c. Provide operating experience with shot-peening of a rotor which demonstrates that compressive stresses are created and increases initiation time for this material and geometry.*

GE began shot-peening nuclear LP rotors approximately 20 years ago. To date, no confirmed (i.e., measurable) SCC cracks have been found in this fleet. General industry opinion about shot-peening and its impact on SCC is reflected in the following statement from Reference 1:

> "...the compressive layer from shot peening removes the tensile stress of the SCC (Venn diagram) triangle. Without tensile stress, SCC failure is significantly retarded or prevented from ever occurring..."

The diagram below, reproduced from Reference 2, demonstrates that compressive stresses are created in GE dovetail geometries by shot-peening.

[[ ]]

References:

1. Shot Peening Applications 9th Ed; Metal Improvement Company 2005 p. 27

2. X-Ray Diffraction Determination of the Residual Stress Distributions in Three NiCrMoV Steel Turbine Wheel Sections; Report #0025-0504 Prepared by Lambda Research Inc.; Cincinnati OH for the General Electric Company, 7/27/1990

**Proposed COLA Revision**

None

**Attachment 1**
**NRC3-11-0028**

**Enclosure 5**

**Response to RAI Letter No. 53**
**(eRAI Tracking No. 5608)**

**RAI Question No. 10.02.03-16**

**NRC RAI 10.02.03-16**

*In a letter dated October 5, 2010, the applicant's response to RAI 4641, Question 10.02.03-11 provided information concerning valve testing. However, the following additional information is requested to clarify the response:*

*a. The RAI response to RAI 4641, Question 10.02.03-11(c) provides a figure (graph) with no scale for the x and y axis on the graph. Please provide the appropriate numbers for the graph. Also, please clarify and discuss the following statement: "The percentage of the updated failure rates that are associated with a valve test frequency of 120 days cannot be determined at this time as there is no data that has been collected with this longer test frequency interval. Assessment of the valve failure data indicates that there are no factors that would prevent the extrapolation of the data to the longer test frequency interval and when assessed against the missile probability analysis the risk resulting from the longer test frequency was considered conservative."*

*b. The RAI response to RAI 4641, Question 10.02.03-11(d) states that no additional data has been collected. Does this statement mean there was no operating experience for these valves after 1984? If there was valve operating experience, confirm that the operating experience since 1984 is bounded by the operating experience before 1984. In other words, is the operating experience prior to 1984 worse than the operating experience after 1984?*

**Supplemental Response**

*a. The RAI response to RAI 4641, Question 10.02.03-11(c) provides a figure (graph) with no scale for the x and y axis on the graph. Please provide the appropriate numbers for the graph. Also, please clarify and discuss the following statement: "The percentage of the updated failure rates that are associated with a valve test frequency of 120 days cannot be determined at this time as there is no data that has been collected with this longer test frequency interval. Assessment of the valve failure data indicates that there are no factors that would prevent the extrapolation of the data to the longer test frequency interval and when assessed against the missile probability analysis the risk resulting from the longer test frequency was considered conservative."*

The maximum recommended valve test interval for the operating fleet of GE nuclear steam turbines remains at 90 days. Despite some evidence that an extension to 120 days may result in maintenance of acceptable annual missile probability for some units, GE has not gathered, nor has any nuclear plant operator submitted to GE, any reliability or failure data for valves tested at 120 day test intervals. Therefore, GE has not made any recommendation that valve test intervals for the existing fleet be extended beyond 90 days. The table below reflects the maximum historical valve test interval recommendations for GE nuclear steam turbines.

[[ ]]

| | Pre-1984 GEK17812 | TIL-969 1984 | TIL-969-3R1 1993 | |
|---|---|---|---|---|
| | | | Built Up | Mono Block |
| Main Stop | Daily | Weekly | Up to 3 Months | 3 Months |
| Control | Weekly | Monthly | Up to 3 Months | 3 Months |
| Intercept/Intermediate | Daily | Weekly | Up to 3 Months | 3 Months |

Section 5.1.2.1 of the ESBWR Steam Turbine Low-Pressure Missile Generation Probability Analysis ST-56834/P provides information regarding the steam valve failure rates used within the analysis. As can be seen within the data set, the extension of the valve test interval from 1984 levels (TIL-969) to 1993 levels (TIL-969-3R1) resulted in no increase in the incidence of valve failures. Data from 1993 and 2008 further indicates that the countermeasures deployed to correct the pre-1993 valve failures were effective in reducing the probability of future failures.

As stated in Section 5.1.2.1 of the ESBWR Steam Turbine Low-Pressure Missile Generation Probability Analysis ST-56834/P, and shown in the graph in RAI response to 10.02.03-16a above, approximately the same level of missile probability risk is realized for a valve test frequency of 120-days (with the updated 2008 valve test failure rates) versus a 90-day test interval with the older valve test failure rates. Thus, GE endorses a 120-day valve test frequency for ESBWR units.

*b. The RAI response to RAI 4641, Question 10.02.03-11(d) states that no additional data has been collected. Does this statement mean there was no operating experience for these valves after 1984? If there was valve operating experience, confirm that the operating experience since 1984 is bounded by the operating experience before 1984. In other words, is the operating experience prior to 1984 worse than the operating experience after 1984?*

RAI 10.02.03-11(d) refers specifically to the hydraulic probability model and failure rates of the hydraulic system. As such, the response was addressing only the hydraulic model and not the valve failure rate model. The valve failure rate data and operating experience is covered in GE-ST report ST-56834/P.

The 1984 hydraulic system reliability model was dominated by two common and known failure modes: (1) water contamination due to leaking EHC oil coolers, and (2) corrosion of non-stainless steel mechanical and/or electrical hydraulic trip valves. Only a small percentage of the existing GE fleet had components that were subject to these failure modes. GE has worked with customers to retrofit affected machines with components of improved design and materials, as well as improving plant maintenance practices (reference: GE TIL 796-2). GE has not gathered, nor has any nuclear plant operator submitted to GE, any formal reliability or failure data for the hydraulic system since the retrofits and operational changes were put into effect. GE has continuously monitored the fleet for reliability or maintenance issues associated with the hydraulic system through our network of inspection services, fleet support, and customer service representatives. To date, no failures or issues within the system have been noted. Thus, the 1984 system reliability model values are considered conservatively bounding for existing units.

The last two paragraphs of GE-ST report ST-56834/P, Section 5.1.4, "Hydraulic Model," discuss the design features of the ESBWR MARK VIe hydraulic system to address the above concerns.

**Proposed COLA Revision**

None