The Detroit Edison Company
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DTE Energy

Detroit Edison

10 CFR 52.79

October 5, 2010
NRC3-10-0045

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. 31 Related to the SRP Sections 10.02.03 for the Fermi 3 Combined License Application," dated April 28, 2010
3) Letter from Richard E. Kingston (GEH) to USNRC, "Transmittal of GE-Energy Steam Turbines (GE-ST) "ESBWR Steam Turbine – Low Pressure Rotor Missile Generation Probability Analysis" ST-56834/P Revision 1 and ST-56834/N-P, Revision 1," dated July 28, 2009
4) Letter from Peter W. Smith to USNRC, "Detroit Edison Company Schedule for Response to NRC Request for Additional Information Letter No. 31," dated July 9, 2010
5) Letter from Richard E. Kingston (GEH) to USNRC, "Transmittal of "ESBWR Steam Turbine – Low Pressure Rotor Missile Generation Probability Analysis" ST-56834/P Revision 2 and ST-56834/N-P, Revision 2," dated September 30, 2010

Subject: Detroit Edison Company Response to NRC Request for Additional Information Letter No. 31

In Reference 2, the NRC requested additional information (RAI) to support the review of certain portions of the Fermi 3 Combined License Application (COLA). The RAIs in Reference 2 requested additional information concerning the "ESBWR Steam Turbine – Low Pressure Rotor Missile Generation Probability Analysis" submitted by General Electric Hitachi (GEH) to the NRC in Reference 3.

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In Reference 4, Detroit Edison committed to providing the responses to the RAIs is Reference 2 14 days after GEH submitted the revised report. Attachment 12 contains proposed COLA Revisions due to GEH submittal of Revision 2 of the "ESBWR Steam Turbine – Low Pressure Rotor Missile Generation Probability Analysis" contained in Reference 5.

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 5th day of October 2010.

Sincerely,

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

Attachments:
1) Response to RAI Letter No. 31 (Question No. 10.02.03-1)
2) Response to RAI Letter No. 31 (Question No. 10.02.03-2)
3) Response to RAI Letter No. 31 (Question No. 10.02.03-3)
4) Response to RAI Letter No. 31 (Question No. 10.02.03-4)
5) Response to RAI Letter No. 31 (Question No. 10.02.03-5)
6) Response to RAI Letter No. 31 (Question No. 10.02.03-6)
7) Response to RAI Letter No. 31 (Question No. 10.02.03-7)
8) Response to RAI Letter No. 31 (Question No. 10.02.03-8)
9) Response to RAI Letter No. 31 (Question No. 10.02.03-9)
10) Response to RAI Letter No. 31 (Question No. 10.02.03-10)
11) Response to RAI Letter No. 31 (Question No. 10.02.03-11)
12) Proposed COLA Revisions

cc:
Adrian Muniz, NRC Fermi 3 Project Manager
Jerry Hale, NRC Fermi 3 Project Manager
Bruce Olson, NRC Fermi 3 Environmental Project Manager
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 Dept. of Natural Resources & Environment
Radiological Protection Section (w/o Attachments)

Attachment 1
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-1

NRC RAI 10.02.03-1

The FERMI , Unit 3, COL FSAR provided information to STD COL 10.2-2-A by referencing the GE-Energy Steam Turbines (GE-ST) report ST-56834/P, Revision 1 as the bounding analysis for the probability of turbine missile generation. Clarify whether the GE-ST report ST-56834/P, Revision 1 is applicable to the GE Model N3R-6F52 turbine.

Response

The General Electric model number (or code type) for the ESBWR steam turbine is the "N3R-6F52" and is based on the LP configuration with a 52" last stage margin bucket in a 3-Low Pressure hood tandem double flow configuration. The code type is incorporated into paragraphs 1.0 and 2.1 of the revised report ST-56834/P.

Attachment 2
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-2

NRC RAI 10.02.03-2

Sections 2.1, 5.1 and 5.1.5 of GE Report ST-56834/P, Revision 1 specify a MARK ™ IVe turbine generator control system (TGCS), while Sections 5.1.6 specifies a MARK ™ VIe TGCS. Clarify whether the ESBWR turbine generator uses a General Electric MARK ™ IVe TGCS or a General Electric MARK ™ VIe TGCS.

Response

The correct nomenclature for the turbine generator control system (TGCS) is the MARK™ VIe. Correction to the nomenclature for the TGCS has been made to sections 2.1, 5.1, and 5.1.5 of the revised report ST-56834/P.

Attachment 3
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-3

NRC RAI 10.02.03-3

Clarify whether the reference of 120 percent of rated speed in Section 2.2 of GE Report ST-56834/P, Revision 1 is the design overspeed of the turbine used in the analysis.

Response

The "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis" report (ST- 56834/P) includes consideration of various potential overspeed events as identified in Table 7.2 and assesses the overall probability of missile generation based on the discrete probability of occurrence for each of the events. The reference to 120% in section 2.2 of the report identifies that the overspeed trip set point and turbine generator control system are designed to prevent the turbine from exceeding 120% of rated speed during an emergency overspeed event.

Attachment 4
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-4

NRC RAI 10.02.03-4

Section 3.1 of GE Report ST-56834/P, Revision 1 states that the low pressure (LP) turbine rotor forgings are NiCrMoV alloy material in accordance with B50A373B8. Provide the General Electric specification B50A373B8 and confirm that this material is used for both existing turbine rotors and the proposed forgings for the ESBWR turbine rotors. Also, provide operational experience of this turbine rotor material, or provide justification for the use of this material if there is no operational experience.

Response

All General Electric solid (i.e., not shrunk-on wheel) nuclear low-pressure (LP) rotors beginning in the 1980's have been manufactured in accordance with GE specification B50A373B8. The proposed LP rotors for ESBWR N3R-6F52 steam turbine will also be manufactured in accordance with specification B50A373B8 or an equivalent specification with more restrictive chemistry requirements. GE specification B50A373B8 may be reviewed by the NRC at the General Electric facility in Schenectady, New York.

Attachment 5
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-5

NRC RAI 10.02.03-5

Section 3.1 .1 of GE Report ST-56834/P, Revision 1 specifies "Deep seated FATT testing, however, will be performed for each production rotor forging during routine material acceptance testing and the fracture toughness at the forging centerline will be derived based on historical correlations." Discuss how the deep seated FATT testing will be performed, including information about the specimen material if it is not from the production rotor forging and the depth of the forging material from where the specimens will be obtained. Present the historical correlations for deriving the fracture toughness at the forging centerline. Justify the use of the historical correlations, since this correlation may not be applicable to the ESBWR rotor forging (e.g., a FATT temperature of 30 °F) due to the different material specification and properties used in the historical data.

Response

Deep-seated FATT testing will utilize the same practices executed on past monoblock rotor forgings. Deep-seated standard sized Charpy V Notch specimens in accordance with ASTM E23 are obtained from each production forging, machined from radial trepans extracted from three axial locations along the main body (largest diameter section) of the rotor forging from material in between the rotor wheels. The radial trepans depths are the maximum depth that can be accommodated by the rotor forging final geometry.

The historical correlation of fracture toughness at the forging centerline is shown in the "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis" report (ST-56834/P) (revised Figure number 3-2). The data is based on historical correlation from NiCrMoV alloy steel forgings manufactured per the requirements of GE material specification B50A373B8. The LP rotor forging material for the ESBWR N3R-6F52 steam turbine will be manufactured in accordance with B50A373B8 or an equivalent specification with more restrictive chemistry requirements. The historical correlation between the centerline fracture toughness and the deep-seated location has been added as Figure 3-1 of report ST-56834/P.

Attachment 6
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-6

NRC RAI 10.02.03-6

Section 3.1 of GE Report ST-56834/P, Revision 1 specifies that the ESBWR low pressure turbine rotor has a FATT temperature of -1.1°C (+30 °F) for a large integral forging. However, Section 3.1.1 states that for missile generation probability calculations, a normally distributed FATT featuring a -30 °F mean and a 30 °F standard deviation is assumed. For a normal distribution, 95% of the sampling will be between the mean value, plus or minus 2σ, i.e., between -90 °F and +30 °F. Demonstrate that the FATT distribution (normally distributed FATT featuring a -30 °F mean and a 30 °F standard deviation) used in the missile generation probability calculation is representative of the material having a FATT value of -1.1°C (+30 °F). In addition, clarify whether the mean FATT value in the turbine missile probability calculations changes as the postulated crack grows.

Response

The FATT values assumed in the "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis" report (ST-56834/P) represent the forging centerline values. FATT values are assumed to be independent of the location in the forging, the properties of the centerline are assumed to exist throughout the forging. The assumption of invariant FATT properties is considered to be conservative based on the correlations that FATT values improve the closer the location is, within the forging, to the quench and temper surface. The design calculations of report ST-56834/P are based on accumulated forging data (11 forgings) from nuclear monoblocks forgings (per B50A373B8) produced within the past 20 years, statistically analyzed. Statistical analysis of the forging data resulted in an average FATT value of -34°F and a plus two-sigma FATT value of +11 °F. The FATT distribution in report ST-56834/P is considered a conservative bounding assumption of the distribution. The mean FATT value does not change as the postulated crack grows, based on the conservative assumption of using centerline FATT values.

Attachment 7
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-7

NRC RAI 10.02.03-7

Confirm that Figure 3-1 of GE Report ST-56834/P, Revision 1 for the NiCrMoV toughness curve is based on actual test samples from General Electric specification B50A373B8 referenced in Section 3.1. Otherwise, provide justification for using the Figure 3-1 curves for the ESBWR turbine missile analysis.

Response

The fracture toughness data curve presented as Figure 3-1 in the "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis" report (ST-56834/P) was developed based on test specimens from monoblocks nuclear rotor forgings produced by General Electric suppliers in accordance with specification B50A373B8.

Attachment 8
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-8

NRC RAI 10.02.03-8

Sections 3.1.3, 4.2.2 and 8.0 of GE Report ST-56834/P, Revision 1 imply that the low pressure turbine rotors can be either bored or solid. Typically, rotors are bored to allow for inspections and to remove material that may have degraded material properties. Therefore, discuss the operating experience of solid rotors, including any effects due to the degraded material properties of the rotor core. Also, discuss the past volumetric in-service inspection (ISI) results and the current ISI technology to establish the minimum flaw size in the turbine rotors that can be detected by the ISI. Compare this minimum flaw size to the average undetected embedded flaw specified in Section 4.2.2.

Response

a.) The existing General Electric nuclear steam turbine LP rotor fleet includes both solid and bored rotors. A high percentage of the older rotors were bored. However, the percentage of bored rotors is much lower in more recent rotor replacements because of more extensive deep-seated volumetric testing (from the outer periphery) and confidence in center core region properties based on the statistical results of testing. The "ESBWR Steam Turbine – Low Pressure Turbine Missile Probability Analysis" report (ST-56834/P) takes into account consideration of deep-seated material properties including the center core region of solid rotors.

b.) As noted in sections 8 and 9 of the "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis" report (ST-56834/P), the calculated annual missile probability is dominated by valve failure and stress corrosion crack in the dovetail region reaching the critical size. The combination of initial forging acceptance test detection capability, cyclic loading profile, and material properties is such that the annual probability of a missile due to an undetected internal forging flaw reaching a critical size before 60 years is less than the NRC specified limit on missile generation probability. As such, in-service volumetric inspection of solid nuclear LP rotors is not required to meet the calculations included in the report.

Attachment 9
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-9

NRC RAI 10.02.03-9

Discuss how the stresses in Section 4.2 of GE Report ST-56834/P, Revision 1 for a solid rotor were derived.

Response

The stresses for both bored and boreless (i.e., solid) rotors were derived using finite element analysis based on the defined wheel geometry for the N3R-6F52 rotor and recommended start-up transient thermal loading. The results of the analysis are presented in Table 4-1 for each stage of the LP steam path. The difference in calculated annual missile probabilities (bored vs. boreless) shown in Figure 9-2 is attributed to the different average tangential stresses shown in Table 4-3 for each type of rotor.

Attachment 10
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-10

NRC RAI 10.02.03-10

Provide the tangential stresses at the slot bottoms of axial entry dovetails in Section 4.3 of GE Report ST-56834/P, Revision 1 and the corresponding stresses around the shrunk-on-wheel keyways for a similar size turbine to demonstrate that the current designs "feature dramatically low tangential stress." Besides low stresses, discuss other reasons which make the use of shrunk-on-wheel crack initiation and growth characteristics in the integral forging application for the GE Model N3R-6F52 turbine conservative.

Response

In recent years, General Electric has derived surface tensile stress limits for nuclear LP rotors (made of GE alloy B50A373B8) below which stress corrosion cracking (SCC) should not statistically occur. These threshold limits are a function of temperature and thus are unique for each stage of the LP turbine and were derived from a combination of proprietary General Electric material tests and fleet experience. The ESBWR LP rotors were designed to avoid tensile surface stresses that exceed the developed threshold surface stress limits.

In stages where the temperature would be prevalent to past observed SCC, the ESBWR N3R-6F52 steam turbine design's surface tensile stress magnitudes are less than previous designs including those featuring shrunk-on wheels. With historical inspections and surface flaw indications from shrunk on wheels and historical dovetail stress calculations together with operational experience, provides an engineering baseline data and capability for extrapolation to new designs.

ESBWR LP rotors will be shot-peened in regions where SCC has been observed in the past. SCC initiation and growth distributions used for ESBWR missile probability analysis are exclusively from non-shot-peened shrunk-on wheels. Because of lower applied tensile stress and the application of controlled shot-peening, the ESBWR LP rotors are less vulnerable to SCC as compared to earlier designs featuring non-shot-peened shrunk-on wheels. The application of shrunk-on wheel SCC characteristics in the "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis" report (ST-56834/P) is therefore, considered to be conservative and historical field inspections confirm this conclusion.

Attachment 11
NRC3-10-0045

Response to RAI Letter No. 31
(eRAI Tracking No. 4641 Revision 2)

RAI Question No. 10.02.03-11

NRC RAI 10.02.03-11

The following is related to Section 5 of GE Report ST-56834/P, Revision 1 which provides an evaluation of the overspeed probability.

a. *Section 5.1.2 states that ESBWR valves are equivalent to those found on the current fleet. The only visible difference is the use of direct actuators on the main steam control and the intercept valves. Provide information, especially from testing and actual operation of direct actuators, demonstrating that the direct actuators are equivalent to nuclear actuators in all design features and functionality.*

b. *Section 5.1 .2.1 states that the steam valve failure rates used for this ESBWR analysis have been updated to include 1993 and 2008 failure data assessment. Provide the 1993 and 2008 failure rates similar to the format provided in the 1984 report, Appendix G.*

c. *Section 5.1 .2.1 states that approximately the same level of missile probability risk is realized for a valve test frequency of 120-days (with the updated failure rates) versus a 90-day test interval with the older failure rates. Substantiate this statement by providing calculated missile probabilities based on different test frequencies. What is the percentage of the updated failure rates that are associated with turbine units with a valve test frequency of 120-days?*

d. *Section 5.1.4 states that the component failure rates used in the hydraulic model are consistent with the values in the 1984 report. Discuss whether these values have been updated with additional data since the 1984 report.*

e. *Section 5.1.6 states that for the MARK ™ VIe TGCS analysis, it was assumed that a load loss occurs once per year. Discuss how this assumption is typical along with any supporting data for this assumption.*

f. *Section 5.1.6 states that the MARK™ VIe TGCS analysis yields a lower overspeed probability than the MARK II TGCS specified in the 1984 report. Provide quantitatively the margin for using the 1984 overspeed probability based on the MARK ™ II TGCS (for existing fleet control and mechanical trip systems) compared to the MARK ™ VIe TGCS analysis overspeed probability result.*

g. *Discuss and compare in tabular form why the valves and TGCS (for the MARK™ II) used in the 1984 report are similar (or more conservative) than the valves and TGCS (for the MARK ™ VIe) used for the ESBWR design, so that it can be concluded that the components are similar so that the failure rates (past operating experience) from the 1984 report (current fleet) can be used for the analysis of the ESBWR design.*

Response

a. *Section 5.1.2 states that ESBWR valves are equivalent to those found on the current fleet. The only visible difference is the use of direct actuators on the main steam control and the intercept valves. Provide information, especially from testing and actual operation of direct actuators, demonstrating that the direct actuators are equivalent to nuclear actuators in all design features and functionality.*

The ESBWR steam turbine main steam control and intercept valve actuators utilize the same functional design and component requirements that General Electric has previously applied to the fossil, nuclear and combined-cycle steam turbine fleets. Previous generation steam turbine control and intercept valve actuator designs were limited by available hydraulic oil operating pressures of that era and required the use of extensive linkage systems to transmit and multiply the available hydraulic forces to the operate the valves. The ESBWR steam turbine system provides a higher pressure (2400 psig) hydraulic power unit that permits elimination of such linkage systems and enable direct mechanical connection of the hydraulic piston/rod/spring unit to the control or intercept steam valve stem for valve operation. The Main Stop Valve and Intercept Stop Valve actuators in use on the existing General Electric nuclear steam turbine fleet are examples of direct actuation designs that would be now applied to the control and intercept valves. Linkage removal offers: reduction in overall number of moving parts, elimination of potential linkage binding and elimination of periodic linkage maintenance (linkage bearing and bushing lubrication). The below table provides a summary of the comparison of actuator function for both the existing fleet and proposed configuration for the ESBWR steam turbine.

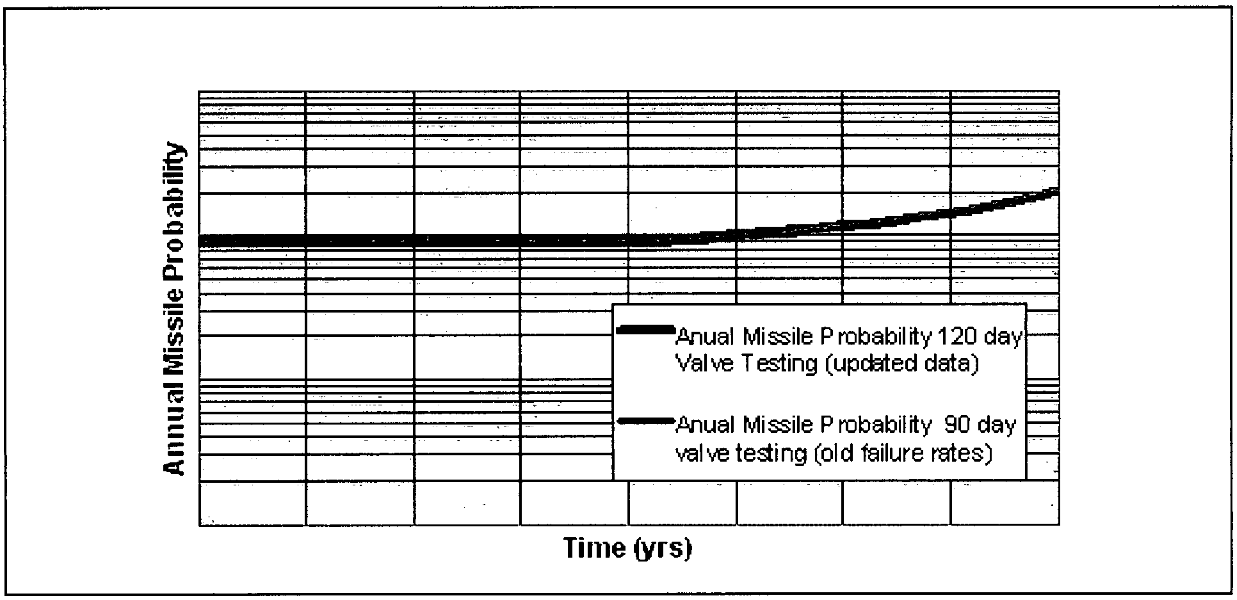
Design Features	**Control and Intercept Actuators – Existing Nuclear Fleet**	**Control and Intercept Actuators – ESBWR**
Failure Position	Fail Closed – Spring Action	Fail Closed – Spring Action
Operating Fluid	Triaryl Phosphate Ester Turbine Oil	Triaryl Phosphate Ester Turbine Oil
Cylinder	Single Acting Hydraulic Piston	Single Acting Hydraulic Piston
Springs	Yes - Helical Coil style	Yes – Belleville style
Closed Position Limit Switch	Yes - IEEE Class 1 E	Yes - IEEE Class 1 E
Servo Valve	4-way, 3 coil (triple redundant coils)	4-way, 3 coil (triple redundant coils)
Fast-Acting Solenoid valve	Yes	Yes
Hydraulic Dump valve	Yes	Yes
Position Transducers	(3)Triple Redundant LVDTs	(3)Triple Redundant LVDTs
Mechanical Force Transmission Method	Four (4) bar linkage (lever force multiplier)	Direct mechanical connection to valve stem

b. *Section 5.1.2.1 states that the steam valve failure rates used for this ESBWR analysis have been updated to include 1993 and 2008 failure data assessment. Provide the 1993 and 2008 failure rates similar to the format provided in the 1984 report, Appendix G.*

Section 5.1.2.1 has been revised to provided the updated valve failure data for the main stop and control valves and the intermediate stop and intercept valves.

c. *Section 5.1.2.1 states that approximately the same level of missile probability risk is realized for a valve test frequency of 120-days (with the updated failure rates) versus a 90- day test interval with the older failure rates. Substantiate this statement by providing calculated missile probabilities based on different test frequencies. What is the percentage of the updated failure rates that are associated with turbine units with a valve test frequency of 120-days?*

The following graph provides the comparison of annual missile probabilities for the 120-day valve test intervals using the updated valve failure data vs. the 90-day valve test intervals using the original 1984 valve failure data. By comparison of the analysis results, the data shows no considerable difference in the missile probability risk for the two test frequencies. 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.



d. *Section 5.1.4 states that the component failure rates used in the hydraulic model are consistent with the values in the 1984 report. Discuss whether these values have been updated with additional data since the 1984 report.*

No additional data has been collected. The failure rates of the 1984 report are considered to be bounding worse case data and are expected to be higher than rates on equipment supplied today given design changes that have been implemented over time to address forced outage events.

e. *Section 5.1.6 states that for the MARK TM VIe TGCS analysis, it was assumed that a load loss occurs once per year. Discuss how this assumption is typical along with any supporting data for this assumption.*

The assumption of the number of load loss events per year was based on early missile probability assessments performed by General Electric dating back to the early 1970's and was used as the basis for the 1984 General Electric Missile Probability Analysis report and all subsequent analyses.

As part of the early assessments a sensitivity evaluation of missile probability was performed on the effects of the assumption for the number of load loss events for three cases:

- 2 events in 3 years,
- 1 loss of load event per year and
- 2 loss of load events per year.

As expected the sensitivity study found that the overall probability increases as the load loss rate increases. The assumed value of one event per year identified in the ESBWR missile probability analysis is considered conservative based on the Turbine Generator design specification provided by GE-H Nuclear, which identified a total of 32 load loss events over the 60-year life of the steam turbine. The turbine Generator design specification uses an estimate of one load loss per refueling (two year period), and based upon the history of nuclear plants in the INPO/WANO databases this rate is conservative.

f. *Section 5.1.6 states that the MARKTM VIe TGCS analysis yields a lower overspeed probability than the MARK II TGCS specified in the 1984 report. Provide quantitatively the margin for using the 1984 overspeed probability based on the MARK TM II TGCS (for existing fleet control and mechanical trip systems) compared to the MARK TM VIe TGCS analysis overspeed probability result.*

Based on an assessment performed by General Electric for the detailed Failure Mode Evaluation Analysis of the MARK™ VIe turbine generator control system, the probability of an overspeed event greater than 120% is seven times less likely with the MARK™ VIe system than previous turbine generator control systems with a mechanical trip mechanism. The details of this assessment may be reviewed at the General Electric facility in Schenectady, New York.

g. *Discuss and compare in tabular form why the valves and TGCS (for the MARKTM II) used in the 1984 report are similar (or more conservative) than the valves and TGCS (for the MARK TM VIe) used for the ESBWR design, so that it can be concluded that the components are similar so that the failure rates (past operating experience) from the 1984 report (current fleet) can be used for the analysis of the ESBWR design.*

Based upon the same logic applied in the GE response to NRC RAI 10.02-03-11, linkage removal offers: reduction in overall number of moving parts, elimination of potential linkage binding and elimination of periodic linkage maintenance (linkage bearing and bushing lubrication).

A summary of the comparison of actuator function for both the existing fleet and proposed configuration for the ESBWR steam turbine is provided in the response to question a above.

Attachment 12
NRC3-10-0045

Proposed COLA Revisions

Markup of Detroit Edison COLA
(following 3 pages)

The following markup represents how Detroit Edison intends to reflect this RAI response in the next submittal of the Fermi 3 COLA. However, the same COLA content may be impacted by revisions to the ESBWR DCD, responses to other COLA RAIs, other COLA changes, plant design changes, editorial or typographical corrections, etc. As a result, the final COLA content that appears in a future submittal may be different than presented here.

Table 1.6-201 Referenced Topical Reports [EF3 SUP 1.6-1]

Report No.	Title	Section No.
NEI 06-13A	Nuclear Energy Institute, "Technical Report on Template for an Industry Training Program Description," NEI 06-13A, Revision 1, March 2008	Appendix 13 BB
NEI 06-14A	Nuclear Energy Institute, "Quality Assurance Program Description," NEI 06-14A, Revision 4, July 2007	17.5
NEI 07-02A	Nuclear Energy Institute, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed under 10 CFR Part 52," NEI 07-02A, March 2008	17.6
NEI 07-03	Nuclear Energy Institute, "Generic FSAR Template Guidance for Radiation Protection Program Description," NEI 07-03, Revision 3, October 2007	Appendix 12 BB
NEI 07-08	Nuclear Energy Institute, "Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)," NEI 07-08, Revision 0, September 2007	Appendix 12 AA
NEI 07-09A	Nuclear Energy Institute, "Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," NEI 07-09A, Revision 0, March 2009	11.5
NEI 07-10A	Nuclear Energy Institute, "Generic FSAR Template Guidance for Process Control Program (PCP)," NEI 07-10A, Revision 0, March 2009	11.4
NEI 06-12	Nuclear Energy Institute, "B.5.b. Phase 2 & 3 Submittal Guideline," NEI 06-12, Revision 3, September 2009	13.6
NEI 08-09	Nuclear Energy Institute, "Cyber Security Plan for Nuclear Power Reactors", NEI 08-09, Revision 3, September 2009	13.6
ST-56834/P	General Electric Company, "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis," ST-56834/P, ~~Revision 1, June 17, 2009~~ Revision 2, September 14, 2010	10.2
ST-56834/N-P	General Electic Company, "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis," ST-56834/N-P, Revision 2, September 14, 2010.	10.2

Chapter 10 Steam and Power Conversion System

10.1 Summary Description

This section of the referenced DCD is incorporated by reference with no departures and/or supplements.

10.2 Turbine Generator

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.2.3.4 Turbine Design

Add the following at the beginning of this section.

STD SUP 10.2-1

The General Electric Company manufactures the turbine and generator. The model N3R-6F52 turbine is from General Electric's N series nuclear steam turbines.

10.2.3.6 Inservice Maintenance and Inspection of Turbine Rotors

Replace the last paragraph with the following.

STD COL 10.2-1-A

The turbine maintenance and inspection program that supports the Original Equipment Manufacturer's turbine missile generation probability calculation is described in DCD Sections 10.2.2.7, 10.2.3.5, 10.2.3.6, and 10.2.3.7. The associated turbine maintenance and inspection frequencies are established in the bounding missile probability analysis in GE-ST, "ESBWR Steam Turbine - Low Pressure Rotor Missile Generator Probability Analysis," ST-56834/P, Revision ~~1~~ $\boxed{2}$, submitted in Reference 10.2-201.

10.2.3.8 Turbine Missile Probability Analysis

Replace the last paragraph with the following.

STD COL 10.2-2-A

The probability of turbine missile generation has been calculated based on bounding material property values in GE-ST, "ESBWR Steam Turbine - Low Pressure Rotor Missile Generator Probability Analysis," ST-56834/P, Revision ~~1~~ $\boxed{2}$, submitted in Reference 10.2-201.

10.2.5 COL Information

10.2-1-A Turbine Maintenance and Inspection Program

STD COL 10.2-1-A This COL item is addressed in Subsection 10.2.3.6

10.2-2-A Turbine Missile Probability Analysis

STD COL 10.2-2-A This COL item is addressed in Subsection 10.2.3.8.

10.2.6 References

MFN-09-484 Supplement 1

10.2-201 GEH Letter, ~~MFN-09-484~~, "Transmittal of ~~GE Energy Steam Turbines (GE-ST)~~ "ESBWR Steam Turbine - Low Pressure Rotor Missile Generation Probability Analysis" ST-56834/P , Revision 2 and ST-56834/N-P, Revision ~~1~~ 2," dated ~~July 28, 2009~~ September 30, 2010

10.3 Turbine Main Steam System

This section of the referenced DCD is incorporated by reference with no departures or supplements.

10.4 Other Features of Steam and Power Conversion System

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.4.5.2.1 General Description

Replace the text with the following.

EF3 CDI The CIRC is depicted in Figure 10.4-201 and Figure 10.4-202. The CIRC consists of the following components:

- Condenser water boxes, piping, and valves
- Condenser tube cleaning equipment
- Water box drain subsystem
- Four 25 percent capacity pumps and pump discharge valves
- A removable assembly of coarse and fine screens that separate the pump forebay (suction) from the cooling tower basin
- One hyperbolic natural draft cooling tower (NDCT)

Table 10.4-3R includes the temperature range of the water delivered by the CIRC pumps to the main condenser.