PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG-17011-NP-A
Revision 2

WESTINGHOUSE NON-PROPRIETARY CLASS 3

Update for Subsequent License Renewal: WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination" and WCAP-15666-A, "Extension of Reactor Coolant Pump Motor Flywheel Examination"

Materials Committee

PA-MSC-1500

October 2019

Westinghouse

PWROG-17011-NP-A
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Update for Subsequent License Renewal: WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination" and WCAP-15666-A, "Extension of Reactor Coolant Pump Motor Flywheel Examination"

PA-MSC-1500

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Risk Applications and Methods - II

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October 2019

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 11, 2019

Mr. W. Anthony Nowinowski
Program Manager
PWR Owners Group,
Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT PWROG-17011-NP, REVISION 2 REPORT, "UPDATE FOR SUBSEQUENT LICENSE RENEWAL: WCAP-14535A, 'TOPICAL REPORT ON REACTOR COOLANT PUMP FLYWHEEL INSPECTION ELIMINATION' AND WCAP-15666-A, 'EXTENSION OF REACTOR COOLANT PUMP MOTOR FLYWHEEL EXAMINATION'" (EPID: L-2018-TOP-0019)

Dear Mr. Nowinowski:

By letter dated May 15, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18143B465), the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) PWROG-17011-NP, Revision 1, "Update for Subsequent License Renewal: WCAP-14535A, 'Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination' and WCAP-15666-A, 'Extension of Reactor Coolant Pump Motor Flywheel Examination,'" to the U.S. Nuclear Regulatory Commission (NRC) staff for review. By letter dated January 28, 2019 (ADAMS Package Accession No. ML19036A684), the PWROG submitted Revision 2 of Topical Report (TR) PWROG-17011-NP to address an NRC Request for Additional Information (Set 5, RAI 4.3.5-2), on the Turkey Point Units 3 and 4 Subsequent License Renewal (ADAMS Accession No. ML18362A146). By letter dated August 6, 2019 (ADAMS Accession No. ML19221B588), the PWROG provided clarification on the intent of the Revision 2 submission and requested that the NRC prepare the final safety evaluation for PWROG-17011-NP against Revision 2.

The NRC staff review determined that the information provided in the TR provides the technical and regulatory basis for continuing the 20-year inservice inspection interval approved for reactor coolant pump motor flywheels for the period of extended operation (60 years) in WCAP-15666-A (ADAMS Accession No. ML18303A413), to the subsequent period of extended operation (80 years). The NRC staff has reviewed PWROG-17011-NP, Revision 2 and concludes that the subject report, as modified by the conditions and limitations summarized in Section 4.0 of the enclosed safety evaluation, provides an acceptable methodology that can be used to support such a request.

By letter dated March 19, 2019 (ADAMS Accession No. ML19072A103), the NRC staff provided the draft safety evaluation (SE) to the PWROG for review and comment. Per email correspondence on June 17, 2019 (ADAMS Accession No. ML19198A037), the PWROG identified that they had no comments on the draft SE.

In accordance with the guidance provided on the NRC website, we request that the PWROG publish an approved version of PWROG-17011-NP, Revision 2 within 3 months of receipt of this letter. The approved version shall incorporate this letter and the enclosed final SE after the title page. Also, it must contain historical review information, including NRC requests for additional information (RAIs) and your responses. The approved version shall include an "-A" (designating approved) following the TR identification symbol. As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and if the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of these TRs, PWROG will be expected to revise the TRs appropriately or justify their continued applicability for subsequent referencing. Licensees referencing these TRs would be expected to justify their continued applicability or evaluate their plant using the revised TRs.

If you have any questions, please contact Jason Drake at 301-415-8378.

Sincerely,

/RA/

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No. 99902037

Enclosure:
Final SE

SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT PWROG-17011-NP, REVISION 2 REPORT, "UPDATE FOR SUBSEQUENT LICENSE RENEWAL: WCAP-14535A, 'TOPICAL REPORT ON REACTOR COOLANT PUMP FLYWHEEL INSPECTION ELIMINATION' AND WCAP-15666-A, 'EXTENSION OF REACTOR COOLANT PUMP MOTOR FLYWHEEL EXAMINATION'" DATE: SEPTEMBER 11, 2019

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ML19198A050 -Package
ML19198A051 -Cover Letter
ML19198A056 -Enclosure
ML19198A037 -Incoming ***concurrence via email** **NRR-106**

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT PWROG-17011-NP, REVISION 2

UPDATE FOR SUBSEQUENT LICENSE RENEWAL: WCAP-14535A, "TOPICAL REPORT ON REACTOR COOLANT PUMP FLYWHEEL INSPECTION ELIMINATION" AND WCAP-15666-A, "EXTENSION OF REACTOR COOLANT PUMP MOTOR FLYWHEEL EXAMINATION"

PRESSURIZED WATER REACTOR OWNERS GROUP

WESTINGHOUSE ELECTRIC COMPANY

1.0 INTRODUCTION AND BACKGROUND

On May 15, 2018, the Pressurized Water Reactor Owners Group (PWROG), formerly Westinghouse Owners Group or WOG, submitted Topical Report (TR) PWROG-17011-NP, Revision 1, "Update for Subsequent License Renewal: WCAP-14535A, 'Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination' and WCAP-15666-A, 'Extension of Reactor Coolant Pump Motor Flywheel Examination'" (Ref. 1), dated May 2018 (referred to as the TR, Revision 1 in the remainder of this document) for the U.S. Nuclear Regulatory Commission (NRC) staff review. Further clarifying information related to this TR, Revision 1 was submitted on August 31 and December 21, 2018 (Refs. 2 and 3), by Florida Power & Light Company (FPL) in response to a request for additional information (RAI) for its subsequent license renewal (SLR) application for Turkey Point, Unit 3 and 4. By letter dated January 28, 2019, the PWROG submitted PWROG-17011-NP, Revision 2 (Ref. 4), to supplement TR, Revision 1 (referred to as the TR in the remainder of this document). The purpose of this TR is to extend the applicability of WCAP-14535A (Ref. 5) and WCAP-15666-A (Ref. 6) to the subsequent period of extended operation (SPEO), i.e., from 60 years of operation to 80 years of operation. The original inspection frequency for reactor coolant pump (RCP) flywheels was specified in Regulatory Guide (RG) 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity," dated August 1975 (Ref. 7).

Prior to 1996, plants gathered more than 20 years of operating experience and inspection results, and there were no service-induced flaws identified which would affect RCP flywheel integrity. Therefore, considering the inspection history, the savings in inspection cost, and the reduction in personnel radiation exposure, the WOG submitted a deterministic and probabilistic fracture mechanics methodology in WCAP-14535A to the NRC in January 1996 for elimination of the RCP flywheel inspection. As indicated in the NRC staff's safety evaluation (SE) of WCAP-14535A, the NRC staff only evaluated the stress and deterministic fracture mechanics part of the methodology and approved the extension of the RCP flywheel inspection interval from 40 months as specified in RG 1.14, Revision 1 to 10 years.

Subsequently in 2001, the WOG submitted and the NRC staff approved an extension of the RCP flywheel inspection interval from 10 years to 20 years for the PEO based on a probabilistic fracture mechanics (PFM) and risk-informed methodology (Ref. 6).

2.0 REGULATORY EVALUATION

The function of the RCP in the reactor coolant system (RCS) of a pressurized water reactor (PWR) plant is to maintain an adequate cooling flow rate by circulating a large volume of primary coolant water through the RCS. Following an assumed loss of power to the RCP motor, the flywheel, in conjunction with the impeller and motor assembly, provides sufficient rotational inertia to assure adequate primary coolant flow during a RCP coastdown, thus maintaining adequate core cooling. A concern regarding the overspeed of the RCP and its potential for failure led to the issuance of RG 1.14. This RG describes a method acceptable to the NRC staff for implementing the requirements of General Design Criterion 4, "Environmental and Missile Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* Part 50, "Licensing of Production and Utilization Facilities."

As mentioned in the introduction, the original inspection interval was specified in RG 1.14, Revision 1. However, following the NRC staff approval for WCAP-14535A and WCAP-15666-A, inspection of RCP flywheels was performed using the approved inspection interval stated in the SEs for WCAP-14535A and WCAP-15666-A (Refs. 5 and 6).

Once approved, the TR will provide the basis to continue a 20-year inspection interval for the RCP flywheels into the SPEO.

3.0 TECHNICAL EVALUATION

3.1 The Stress and Deterministic Fracture Mechanics Methodology Evaluation

The primary regulatory position of RG 1.14, Revision 1, regarding flywheel design concerns three critical speeds: (a) the critical speed for ductile failure, (b) the critical speed for non-ductile failure, and (c) the critical speed for excessive deformation of the flywheel. This regulatory position specifies, as a design criterion, that the normal speed of the flywheel should be less than one-half of the lowest of the critical speeds, and the loss-of-coolant accident (LOCA) overspeed should be less than the lowest of these three critical speeds.

Section 2.3 of this TR documents the stress and fracture mechanics analyses that were performed for two bounding RCP flywheel groups to address these RG issues. They are identical to those in WCAP-14535A (WCAP-14535A contains analysis results for the complete RCP flywheel groups) and WCAP-15666-A. Since fatigue crack growth (FCG) is the only time-limited aging mechanism and the total cycle assumption of 6000 in WCAP-14535A is very conservative, this cycle assumption was used in WCAP-15666-A and again in this TR, considering operating experience. Consequently, the NRC staff's acceptance of the WOG's evaluation in Reference 5 regarding the following to meet the RG 1.14, Revision 1, requirements remains applicable to this TR:

- Material Information
- Analysis for Critical Speed Based on Ductile Fracture
- Analysis for Critical Speed Based on Non-ductile Failure
- Compliance with the Excessive Deformation Failure Criterion
- Compliance with LOCA Overspeed Criterion

3.2 The PFM Methodology Evaluation

Important results from the deterministic fracture mechanics analysis show that the critical crack length is 3.1 inches for Group 1 RCP flywheels under the lowest fracture toughness assumption, and the crack growth after 80 years is 0.08 inch. The relatively small crack growth could be used to qualitatively justify the continued adoption of the 20-year inspection interval in the SPEO. However, Westinghouse supplements the deterministic evaluation with a qualitative and quantitative risk assessment using a methodology consistent with RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis" (Ref. 8).

Analytically, the failure of the flywheel is determined by the presence of a crack large enough that, if subjected to the stresses caused by a given speed, will cause the flywheel to fail. The length of a flaw that will cause failure is defined as the critical flaw size, which depends on the rotating speed of the flywheel and the fracture toughness of the flywheel material. Given an event with a specific flywheel speed, if a crack grows from an assumed initial flaw size to the critical size during the evaluation time, then the flywheel is assumed to fail. Since all parameters mentioned above have some uncertainty, this TR treats them as random variables with distributions and uses a PFM methodology to estimate the conditional probability of failure (PoF) of the flywheel for the event. These conditional PoFs for RCP flywheels are essential inputs to the risk assessment.

The PFM methodology provided in this TR employed the core Monte-Carlo simulation modules with importance sampling that have been adopted and validated in the Structural Reliability and Risk Assessment (SRRA) model, supporting WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report" (Ref. 9). Due to the simplicity of the RCP flywheel geometry and loading, only six flywheel-specific equations needed to be incorporated into the standard subroutines of the computer code to support PFM analyses in this TR. The limited number of equations needing modification also made identification and correction of any errors in coding easier. Therefore, evaluation of the PFM methodology for RCP flywheels is reduced to evaluation of the input constants and random variables (with mean and standard deviation (σ) for a normal distribution or a median and factor for a log-normal distribution).

3.2.1 The Driving Force for an Assumed Crack

Regarding the driving force for an assumed crack, the loading in the PFM methodology is reflected in two normally distributed random variables: (1) speed change per transient and (2) speed for design limiting events. The first random variable corresponds to the normal speed with a mean of 1200 rpm and a σ of 120 rpm for FCG calculations. The NRC staff found this acceptable because using the steady state condition parameter (in this case, the normal speed) as a mean value in the FCG calculation is a standard approach by the industry and the NRC. Further, the assumption that σ is 10 percent of the mean is commonly used when actual data are not available for a statistical analysis. The second random variable corresponds to the design limiting speed with a mean of 1500 rpm and a σ of 150 rpm for the first three events identified in Table 3-7 of the TR and a mean of 3321 rpm for the fourth event identified in the table. The NRC staff approved 1500 rpm as the design speed and 3321 rpm as the peak speed for the double ended guillotine break (DEGB) LOCA event in WCAP-15666-A (Ref.6). This acceptance remains valid during the SPEO because these speeds are not time dependent. Likewise, using 10 percent of the mean value as the σ value for the limiting event speed is acceptable as stated above. The corresponding speeds for Calvert Cliffs are 1125 rpm and

1368 rpm. Since the TR is not responsible for the accuracy of its reported plant-specific data, these speeds should be confirmed in the future Calvert Cliffs SLR application. This is further discussed in Section 4.0.

The randomly selected speed, when combined with a randomly selected initial flaw length, adjusted for crack growth based on a randomly selected crack growth rate (CGR), would determine the change of stress intensity factor (ΔK) per transient for the fatigue crack growth evaluation and the applied K for the critical flaw evaluation. This TR used a log-normally distributed initial flaw length with a median of 0.1 inch and an uncertainty factor of 2.153. These are acceptable to the NRC staff because they are more conservative than the values used in the SRRA program, supporting WCAP-14572, Revision 1-NP-A.

For FCG calculations, the following CGR equation is used,

$$da/dN = C_0(\Delta K)^n,$$

where C_0 is treated as a log-normally distributed random variable with a median of $9.95\text{x}10^{-11}$ and an uncertainty factor of 1.414. The exponent "n" is treated as a constant with a value of 3.07, consistent with the ASME Code FCG equation for ferritic steels in an air environment. The ASME Code FCG equation represents best-estimate values. However, the TR used one half of the C_0 in the ASME Code FCG equation as the median for this random variable and, thus, underestimated the CGR by 50 percent (correct CGR = 2 x TR CGR). The NRC staff evaluated and found that this non-conservatism is completely compensated for by considering the shrink-fit stresses, which were neglected in the TR CGR calculations. The response to an RAI documented in WCAP-14535A (Ref. 5) indicated that, if the shrink-fit stresses were considered, the ΔK and the CGR would be significantly reduced due to decreased stresses at high speed caused by reduced shrink-fit and increased stresses at zero speed caused by full shrink-fit. Therefore, using the TR CGR would overestimate the CGR by 150 percent (correct CGR = 0.4 x TR CGR). The NRC staff found that the TR CGR is acceptable because the net effect of underestimating C_0 and overestimating ΔK will result in an overestimation of the CGR by 25 percent (correct CGR = 2 x 0.4 TR CGR). Consequently, the NRC staff determined that the FCG methodology that was used to generate the conditional PoF values presented in Table 3-3 and Tables 3-6 to 3-9 of the TR is acceptable.

Another random variable related to FCG is the number of transients per operating cycle. The TR used a mean of 100 transients and a σ of 10 transients for this random variable. This is very conservative and acceptable because a plant is unlikely to experience 100 transients in an operating cycle which would require frequent RCP starts and stops.

3.2.2 Fracture Resistance for an Assumed Crack

Regarding the fracture resistance of the flywheel to a crack after 80 years of growth, the PFM methodology determines the K_{IC} value using two normally distributed random variables: (1) operating temperature of the flywheels and (2) nil-ductility transition reference temperature (RT_{NDT}) value of the flywheel material. The K_{IC} value can then be determined by these two variables using the ASME Code Appendix A K_{IC} equation. The TR used a mean of 95 °F for Westinghouse plants (70 °F for Calvert Cliffs) and a σ of 12.5 °F for the flywheel operating temperature. The TR states that the containment building temperature is typically 100 °F to 120 °F. Therefore, using the above mean and σ for the flywheel ambient temperature would be conservative and acceptable because (1) 95 percent of the time, the randomly selected ambient temperature would be between 70°F (mean - 2σ) and 120°F (mean + 2σ) for Westinghouse

plants, giving lower K_{IC} values for the biased lower temperatures; and (2) the corresponding 95 percent ambient temperature range is between 45 °F and 95°F for Calvert Cliffs, giving an even lower K_{IC} values. The TR used a mean of 30 °F and a σ of 17 °F for the RT_{NDT} value. Using these values are conservative, considering that the first flywheel specification dated 1969 requires the RT_{NDT} values from both longitudinal and transverse Charpy specimens be less than 10 °F. To supplement this determination, the NRC staff examined the initial RT_{NDT} values for SA-533 Grade B material for reactor pressure vessels (RPVs) fabricated before 1975 (approximately the same vintage) and found the highest initial RT_{NDT} value for this RPV material is 60 °F, supporting the TR's selection of mean and σ for the RT_{NDT} value.

The TR provides no description for the crack initiation toughness factor (F-KIC), which is another random variable with a mean value of 1.0 and a σ of 0.1. In the PFM analysis, the fracture toughness K_{IC} can be determined by the randomly selected RT_{NDT} value and the operating temperature of the flywheel. Therefore, F-KIC must be related to the use of the ASME Code K_{IC} curve. Since the ASME Code K_{IC} curve is a 95 percent lower bound curve, applying F-KIC to it, or a mean curve based on the ASME Code K_{IC} curve, is acceptable. Reference 3 confirmed that the TR used the mean K_{IC} curve.

3.2.3 Constant Variables Related to Inspections

Except for the flywheel inner radius, outer radius, and the FCG exponent, all other constant variables are related to inspections. The TR used 0.1 for the probability of a flaw existing after the preservice inspection. This is conservative because the preservice inspection was performed to detect and to repair all relevant indications and it is unlikely to miss a flaw emanating from the keyway which is 0.1 inch radially and 6.5 inches through-thickness.

The TR used 3 for the operating cycles for the first ISI and 4 for the operating cycles between ISIs. Since Table 3-3 of the TR used "ISI at 4-year intervals" to characterize the PFM results, it is clear that the operating cycle is conservatively set to one year in the PFM analyses. Using these ISI parameters for the first 10 years of operation is consistent with RG 1.14, Revision 1. The risk increase based on the PFM results between the base case having ISIs at the entire 80 years (20 ISIs) and the proposed case having ISIs for only the first 10 years (about 3 ISIs) is evaluated in the risk assessment in Section 3.3 of the TR. This assumption is very conservative because it maximized the difference in the number of ISIs between the two cases. Based on this, the NRC staff determined that there is additional margin in the conditional PoF difference between these two cases.

In summary, the NRC staff determined that the PFM methodology that was used to generate the conditional PoF values presented in Table 3-3 and Tables 3-6 to 3-9 of the TR is acceptable.

3.3 Risk Assessment

The quantitative risk assessment discussed in the TR provides the justification for applying the WCAP-15666-A 20-year flywheel inspection interval for 80 years of operation. Specifically, the risk analyses confirm that applying the inspection extension to flywheels in operation up to 80 years has a negligible impact on risk (core damage frequency (CDF) and large early release frequency (LERF)), i.e., it is within the risk acceptance criteria of RG 1.174. Section 3 of the TR provides a discussion on the requirements of RG 1.174, and extends the previous flywheel failure probability assessment in WCAP-15666-A to 80 years of operation.

The risk evaluation includes the likelihood that a crack will grow large enough to cause failure at the following conditions or events: (1) normal plant operation resulting in a plant trip, (2) a transient or LOCA event with no loss of electrical power to the RCP, (3) a transient or LOCA event with loss of electric power to the RCP motor, and (4) after the DEGB LOCA with the simultaneous loss of power to the RCP motor.

Section 3.4.3 of the TR provides descriptions and frequencies of the initiating events for the different conditions listed in Table 3-5. The NRC staff noted apparent inconsistencies between TR Tables 3-5, 3-7, and 3-8 and WCAP-15666-A. Tables 3-12 and 3-13 regarding conditions for LOCA events versus non-LOCA events. In the RAI response to a SLR application for Turkey Point Units 3 and 4 (Ref. 2), FPL clarified that the two documents represent the same conditions and employ the same analysis assumptions. Furthermore, it clarified that all entries are bounded by the design limiting transient. The NRC staff finds this clarification acceptable and no changes are needed to further clarify information in the TR.

The risk assessment requires an estimate of the conditional PoF of the flywheel at a given event (speed), an estimate of the event frequency, and the change in CDF and LERF given a flywheel failure.

3.3.1 Flywheel Conditional Failure Probability

The method for calculating flywheel failure probabilities is based on the method described in section 3.3.1 of the TR and evaluated in Section 3.2 of this SE. In the evaluation, the NRC staff also relied on the more detailed information in WCAP-15666-A.

3.3.2 Core Damage Evaluation

The failure of the RCP motor flywheel during normal plant operation would directly result in a reactor trip. However, the potential indirect or spatial effects associated with a postulated flywheel failure present a greater challenge in terms of failure effects or consequences. The flywheel has the potential to catastrophically fail, resulting in flywheel fragments, which are essentially high energy missiles, which could impact other structures, systems, and components (SSCs) important to plant safety. Failure of these other SSCs could potentially impact the overall plant safety in terms of core damage (e.g., as a result of the loss of safety injection) or large early release (as a result of potential impacts on containment structures or systems).

In order to address plant specific design differences on a generic basis, the risk assessment in the TR conservatively assumed that failure of the RCP motor flywheel results in core damage and a large early release, i.e., the flywheel failure frequency is equal to CDF and LERF. Section 3.3 of the TR discusses the process for estimating the likelihood of the primary failure mode of the RCP motor flywheel. Section 3.4 of the TR then combines this failure probability estimation with the likelihood of various plant events and consequences to estimate the change in risk for continuing the 20-year examination interval for RCP flywheels in the SPEO.

To investigate the consequences of RCP overspeed, the TR analyzed a spectrum of LOCA events resulting in a range of flywheel transients. Results of this analysis indicated that the limiting event was the DEGB with an instantaneous loss of power, this led to a peak flywheel speed of 3321 rpm. It was also noted that the 3 ft^2 break area case showed a decrease in speed such that the normal operating speed is not exceeded. Based on the WCAP-15666-A assessments, the following scenarios are associated with the primary mode of potential failure

in the Westinghouse RCP motor flywheel that are related to operating speed and potential overspeed during various conditions:

- Failure during normal plant operation resulting in a plant trip (1200 rpm peak speed)
- Failure of the RCP motor flywheel associated with a plant transient or LOCA event with no loss of electrical power to the RCP (1200 rpm peak speed)
- Failure of the RCP motor flywheel associated with a plant transient or LOCA event up to 3 ft^2 with an instantaneous loss of electrical power to the RCP (1200 rpm peak speed)
- Failure of the RCP motor flywheel associated with a DEGB coincident with an instantaneous loss of electrical power, such as loss offsite power (3321 rpm peak speed). This case bounds and is conservatively applied to all flywheel transients for LOCA break areas.

The TR states that the normal operating speed of the flywheel is 1189 rpm with a synchronous speed of 1200 rpm. If there is a pipe rupture in the RCP's outlet piping, the high reactor coolant pressure will force reactor coolant out through the RCP into the low pressure containment structure and hydraulic torque would be applied to the shaft in the direction of increasing shaft speed. If electrical power is maintained to the RCP motor, the motor will function as a dynamic brake and limit the increase in speed of the shaft to less than 1500 rpm. If, however, electric power is lost to the RCP, the flywheel will accelerate. The maximum estimated flywheel speed is 3321 rpm for a DEGB, the largest possible break in the RCP outlet piping with simultaneous loss of electric power to the RCP motor.

The TR evaluated the likelihood of a flywheel developing a critical crack for the overspeed of 3321 rpm and the overspeed of 1500 rpm. The speed of 3321 rpm is used for the DEGB with loss of electric power to the RCP motor (Event 4). The speed of 1500 rpm is used for all other scenarios (Events 1 to 3). The probability of cracks reaching the critical sizes given different inservice inspection (ISI) programs was estimated using the PFM methodology discussed in Section 3.2. In order to develop estimates applicable to the current fleet of reactors, the TR conservatively estimated the impact of the requested 20-year inspection interval by assuming that, aside from the initial inspections during the first 10 years of plant life, there will be no more inspections over the operating life of the units. The base case for comparison assumed ISIs for the entire 80 years. As discussed in Section 3.2.3, this conservative evaluation is acceptable. Given these four events, the conditional PoFs of the flywheel from the PFM results are presented in Table 3-3 of the TR.

The staff noted that the PoF values in Table 3-3 of the TR, Revision 1 for the Groups 1 and 2 flywheels are nearly identical to those from WCAP-15666-A for 60 years of operation, but slightly lower. This could be reasonable because the PFM analyses used 100 fatigue cycles per year. When ISI was performed every 4 years for 60 and 80 years, the total number of ISIs is 15 and 20, respectively. The slightly lower PoF for 80 years reflects the benefit from 5 additional ISIs. However, the NRC staff also found that the TR, Revision 1 needs to explain why the PoFs for the case with ISI performed every 4 years for only the first 10 years in 80 years of operation are lower than the corresponding PoFs in WCAP-15666-A for 60 years. Reference 3 indicated that through a verification and validation process, Westinghouse found that changes were made to the core PFM executable program that produced PoF results in the TR. These changes were

not managed through configuration control and resulted in calculations that should be discarded. The Westinghouse effort also discovered a mistake in the mean K_{IC} equation in the PFM calculation. Therefore, Westinghouse provided revised PoF values to replace the values in Table 3-3 of the TR, Revision 1. The NRC staff reviewed Reference 3 and considers the explanation of the loss of configuration control and the discovery of a mistake in the core PFM program credible and the corrective actions appropriate. Therefore, the revised PoF values can be used in the subsequent risk evaluation. The PWROG later issued Revision 2 of the TR in January 2019 based on the new information and the updated PoF values in Reference 3. These updated PoF values are reported in Table 3-3 of Revision 2 of the TR. As stated in Section 1.0 of this SE, Revision 2 of the TR is referred to simply as "the TR."

The peak speed of the flywheel during normal operation (Event 1) is about 1200 rpm. Because the PFM results shown in Figures 3-6 to 3-8 of Reference 5 indicated that the failure probability of the RCP flywheels is not changing much with time, the PoF under normal operation during any given year can be reasonably approximated by the cumulative probability at the end of 80 divided by 80 years.

The failure of a flywheel is an irreversible change of state, i.e., the flywheel cannot be repaired and returned to service and there is no likelihood that more than one flywheel failure may occur during the operating lifetime. Without the possibility of multiple failures during the operating lifetime, the assumption – that the probability that a critical flaw exists prior to each transient is equal to the probability that a critical flaw will develop by the end of the 80 year operating life – is conservative and acceptable.

Rather than attempt to develop a full spectrum of break sizes, overspeeds, and associated critical sizes, Table 3-5 in the TR stated that LOCAs up to a three square foot break in the primary loop with a loss of electric power to the RCPs would not cause the flywheel to exceed 1500 RPM. Therefore, the frequency of LOCAs up to this size are grouped together with the transient frequency.

The TR estimated the frequency of large break LOCA (LBLOCA) events with break areas in excess of three square foot break using NUREG-1829 and, therefore, the NRC staff finds that the LOCA frequency is appropriate for use in support of this submittal.

A DEGB must be accompanied by a simultaneous loss of power to the RCP motor in order for the flywheel speed to exceed 1200 rpm. Loss of power to the RCP motor is most likely caused by a loss of station power caused by transfer from the offsite electrical grid to the onsite emergency electrical grid, and failure of the emergency grid to properly load and operate. The probability of the loss of station power is dependent on the LOCA because the changing electrical configuration and loads induced by the LOCA may cause the loss of power. Evaluation of the potential for loss of station power indicates that a reasonable estimate for the probability of loss of station power following a LOCA is about 1.4E-2 (Ref. 10). The TR uses this conditional probability.

The above discussion regarding Westinghouse RCP motor flywheels applies to Calvert Cliffs flywheels also, but with the reduced overspeeds specific to Calvert Cliffs (i.e., 1125 RPM and 1368 RPM).

The NRC staff noted, however, that Tables 3-7 to 3-9 of the TR show that the event frequency for the fourth condition is 1.4E-8/year. In WCAP-15666-A, Table 3-12, the corresponding event frequency is 2.8E-8/year based on a maximum LOCA frequency (LOCAs with greater than

5000 gpm blowdown) of 2E-6/year and the probability of loss of station power following a LOCA of 1.4E-2/year. In the RAI response to the SLR application for Turkey Point, Units 3 and 4 (Ref. 2), FPL indicated that the frequencies of LBLOCA events with break areas in excess of 3 ft^2 (fourth condition) reported in WCAP-15666-A were based on Westinghouse fracture mechanics calculations performed prior to NRC issuance in 2008 of NUREG-1829 (Ref. 11). The frequencies of LBLOCA events with break areas in excess of 3 ft^2 estimated in the TR were updated based on NUREG-1829 and the mean failure rates associated with the larger LOCA break sizes presented in Table 7.19 of NUREG-1829. NUREG-1829 states that, "The results in Table 7.19 are appropriate to use for PRA [probabilistic risk assessment] applications that separately consider SGTRs [steam generator tube ruptures]." Specifically, in establishing the large break frequency, the 14 inch and 31 inch diameter breaks were extrapolated to 80 years and interpolated to determine a cumulative frequency for a 3 ft^2 break. The NRC staff finds the use of NUREG-1829 acceptable because it is consistent with current probabilistic risk assessment practices and provides LOCA frequencies as a function of break size.

Consequence Estimate

The flywheel has the potential to catastrophically fail, resulting in flywheel fragments which are essentially high energy missiles that could impact other SSCs important to plant safety. The TR states that the initial investigations indicate that there is not much uniformity with respect to the layout of critical targets that potential flywheel fragments could impact given its failure and, therefore, a generic damage scenario is difficult to develop. The TR assumes that a flywheel failure would lead directly to core damage and large early release. Therefore, the adequacy of a generic scenario and the quality of the probabilistic risk assessment analysis used to support the TR methodology are not issues and the consequence evaluation is acceptable.

Risk Estimates

The detailed results for all the scenarios for each of the three flywheel types are provided in Tables 3-7 to 3-9 of the TR. The NRC staff performed independent calculations to verify the accuracy of these updated risk values and listed the staff's calculated values in the following Table along with the TR values for comparison. The Table shows that the discrepancy between the NRC staff's and the licensee's risk value is within 0.2 %, representing a very good match. However, for a very small risk value such as the current case, an insignificant discrepancy of 0.2 % between the NRC staff's and the licensee's risk value could cause a discrepancy as large as 13 % between the NRC staff's and the licensee's increase in risk value (i.e., Δ risk).

	Group 1		Group 2	
	NRC	TR	NRC	TR
CDF and LERF with ISI after 10 years	2.125E-08	2.12E-08	1.317E-08	1.31E-08

CDF and LERF without ISI after 10 years	2.156E-08	2.16E-08	1.430E-08	1.43E-8
Increase in CDF and LERF for one flywheel	3.10E-10	3.57E-10	1.13E-09	1.19E-9
Increase in CDF and LERF for four flywheels	1.24E-09	1.43E-09	4.52E-09	4.75E-9

* CDF values for Groups 1 and 2 are based on PoF values from Reference 3, which are identical to those in the TR.

As can be seen in the table, the bounding estimated increase in risk is 4.52E-09/year for Group 2 flywheels. This estimate is well below the very small change in LERF guideline of 1E-7/year in RG 1.174. The NRC staff noted that the revised CDF values for Groups 1 and 2 based on PoF values from Reference 3 are less than the corresponding values in Reference 1. This indicated that if the revised PFM program in Reference 3 is used to generate the CDF values for Calvert Cliffs, the CDF values will also be less than those in Reference 1, establishing that the estimate for Calvert Cliffs is also well below the very small change in LERF guideline of 1E-7/year in RG 1.174. The updated risk increase for Calvert Cliffs in the TR supports the above qualitative justification.

4.0 LIMITATIONS AND CONDITIONS

There is no limitation or condition for the Westinghouse RCP flywheels, considering that the flywheel operating and material data used in the generic analyses have already been examined twice and accepted during plant-specific applications using WCAP-14535A and WCAP-15666-A. However, the flywheel operating and material data used in the generic analyses for Calvert Cliffs in this TR are new and need to be confirmed by the licensee:

- the normal operating speed for the RCP flywheels is 900 rpm
- the design limiting speed for the RCP flywheels is 1125 rpm
- the maximum overspeed following a design basis LOCA is 1368 rpm
- it is appropriate to use 70 °F as the medium temperature for design limiting event (Table 3-2) in the PFM analysis

The TR requires applicants of this TR to confirm that 6000 cycles for 80 years of operation is applicable on a plant-specific basis. This confirmation shall be made in all SLR applications to fulfill the TR requirement. Please note that TR requirements are not considered as SE limitations and conditions.

5.0 CONCLUSIONS

The change in risk estimate includes numerous conservative assumptions including:

- The use of the 1500 rpm for 1189 and 1200 rpm scenarios for Westinghouse RCP flywheels and 1125 rpm for 900 rpm scenarios for Calvert Cliffs.
- The use of the probability that a critical crack exists at the end of the 80 year life as the probability that the crack would exist during each operating year.
- The use of 100 start-ups and shutdowns per calendar year when simulating the FCG.
- Maximizing the difference in number of ISIs between the base case and the proposed case.
- Characterizing the DEGB flow rate as 5000 gpm or higher.
- The failure of the flywheel will cause core damage and a large early release event with a probability of 1.

Considering all the conservative assumptions listed above, the NRC staff determined that the increase in CDF and LERF values for three groups of flywheels provide a bounding estimate of the change in risk associated with the continued adoption of the 20-year inspection interval from the PEO to the SPEO. The bounding estimate is below the very small change in LERF guidelines in RG 1.174, and the NRC staff finds that the increase in risk is small and is consistent with the Commission's Safety Goal Policy Statement.

The TR also addresses the other key principles of risk-informed licensing actions described in RG 1.174. There are no changes to the evaluation of design basis accidents and the margin of safety is being maintained. Nondestructive examinations and inspections will continue to be conducted every 20 years. Therefore, the NRC staff finds the requested change to be well-defined, consistent with defense-in-depth philosophy, contains adequate margin of safety, and incorporates a performance measurement strategy to monitor the change. The NRC staff also finds that the risk evaluation is consistent with the risk-informed methodology and guidelines described in RG 1.174 and that the potential change in risk caused by the continued adoption of the 20-year inspection interval from the PEO to the SPEO is small and acceptable.

The request is a change from the current RG 1.14, Revision 1 guidance. The NRC staff finds that the regulatory positions in RG 1.14, Revision 1 concerning the three critical speeds are satisfied, and that the TR evaluation indicating that critical crack sizes are not expected to be attained during the 20-year inspection interval in the SPEO is reasonable and acceptable. The potential for failure of the RCP flywheel is, and will continue to be, negligible during normal and accident conditions.

6.0 REFERENCES

1. Letter from Ken Schrader, Pressurized Water Reactor Owners Group, to USNRC, May 15, 2018, enclosing Westinghouse Electric Company Report PWROG-17011-NP, Rev. 1, "Update for Subsequent License Renewal: WCAP-14535A, 'Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,' and WCAP-15666-A, 'Extension of Reactor Coolant Pump Motor Flywheel Examination'" Non-Proprietary Class 3, dated May 2018 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML18143B465).

2. Letter from William Maher, Florida Power & Light Company, to USNRC, August 31, 2018, enclosing Florida Power & Light Company response to NRC request for additional information Set 1 regarding Turkey Point Units 3 and 4 subsequent license renewal application (ADAMS Accession No. ML18248A257).

3. Letter from William Maher, Florida Power & Light Company, to USNRC, December 21, 2018, enclosing Florida Power & Light Company response to NRC request for additional information Set 5 Response 4.3.5-2 Revision regarding Turkey Point Units 3 and 4 subsequent license renewal application (ADAMS Accession No. ML18362A146).

4. Letter from Ken Schrader, Pressurized Water Reactor Owners Group, to USNRC, January 28, 2019 enclosing Westinghouse Electric Company Report PWROG-17011-NP, Rev. 2, "Update for Subsequent License Renewal WCAP-14535A, 'Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,' and WCAP-15666-A, 'Extension of Reactor Coolant Pump Motor Flywheel Examination'" Non-Proprietary Class 3, dated May 2018 (ADAMS Package Accession No. ML19036A684).

5. WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," November 1996 (ADAMS Accession No. ML18312A151).

6. WCAP-15666-A, "Extension of Reactor Coolant Pump Motor Flywheel Examination," October 2003 (ADAMS Accession No. ML18303A413).

7. U.S. Nuclear Regulatory Commission, "Reactor Coolant Pump Flywheel Integrity," Regulatory Guide 1.14, Revision 1, August 1975 (ADAMS Accession No. ML003739936).

8. U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Revision 3, January 2018 (ADAMS Accession No. ML17317A256).

9. WCAP-14572, Revision 1-NP-A, *"Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report,"* February 1999 (ADAMS Accession No. ML042610469).

10. NUREG/CR-6538, "*Evaluation of LOCA with Delayed LOOP and LOOP with Delayed LOCA Accident Scenarios*," July 1997 (ADAMS Accession No. ML071630062).

11. NUREG-1829, "Estimating LOCA Frequencies Through the Elicitation Process," April 2008 (ADAMS Accession Nos. ML081060300 and ML082250436).

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Date: July 25, 2019

ACKNOWLEDGEMENTS

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United States Member Participation* for PA-MSC-1500

Utility Member	Plant Site(s)	Participant Yes	Participant No
Ameren Missouri	Callaway (W)	X	
American Electric Power	D.C. Cook 1 & 2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)		X
Dominion Connecticut	Millstone 2 (CE)		X
	Millstone 3 (W)	X	
Dominion VA	North Anna 1 & 2 (W)	X	
	Surry 1 & 2 (W)	X	
Duke Energy Carolinas	Catawba 1 & 2 (W)	X	
	McGuire 1 & 2 (W)	X	
	Oconee 1, 2, & 3 (B&W)	X	
Duke Energy Progress	Robinson 2 (W)	X	
	Shearon Harris (W)	X	
Entergy Palisades	Palisades (CE)		X
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)		X
Entergy Operations South	Arkansas 1 (B&W)		X
	Arkansas 2 (CE)		X
	Waterford 3 (CE)		X
Exelon Generation Co. LLC	Braidwood 1 & 2 (W)	X	
	Byron 1 & 2 (W)	X	
	TMI 1 (B&W)		X
	Calvert Cliffs 1 & 2 (CE)	X	
	Ginna (W)		X
FirstEnergy Nuclear Operating Co.	Beaver Valley 1 & 2 (W)		X
	Davis-Besse (B&W)		X
Florida Power & Light \ NextEra	St. Lucie 1 & 2 (CE)		X
	Turkey Point 3 & 4 (W)	X	
	Seabrook (W)	X	
	Pt Beach 1 & 2 (W)	X	
Luminant Power	Comanche Peak 1 & 2 (W)		X
Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)		X
PSEG – Nuclear	Salem 1 & 2 (W)		X
South Carolina Electric & Gas	V.C. Summer (W)	X	
So. Texas Project Nuclear Operating Co	South Texas Project 1 & 2 (W)		X
Southern Nuclear Operating Co.	Farley 1 & 2 (W)		X
	Vogtle 1 & 2 (W)		X
Tennessee Valley Authority	Sequoyah 1 & 2 (W)		X
	Watts Bar 1 & 2 (W)		X
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)		X
Xcel Energy	Prairie Island 1 & 2 (W)	X	

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PWR Owners Group
International Member Participation* for PA-MSC-1500

Utility Member	Plant Site(s)	Participant Yes	Participant No
Asociación Nuclear Ascó-Vandellòs	Asco 1 & 2 (W)		X
	Vandellos 2 (W)		X
Axpo AG	Beznau 1 & 2 (W)		X
Centrales Nucleares Almaraz-Trillo	Almaraz 1 & 2 (W)		X
EDF Energy	Sizewell B (W)		X
Electrabel	Doel 1, 2 & 4 (W)		X
	Tihange 1 & 3 (W)		X
Electricite de France	58 Units		X
Eletronuclear-Eletrobras	Angra 1 (W)		X
Emirates Nuclear Energy Corporation	Barakah 1 & 2		X
EPZ	Borssele		X
Eskom	Koeberg 1 & 2 (W)		X
Hokkaido	Tomari 1, 2 & 3 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co , LTD	Mihama 3 (W)		X
	Ohi 1, 2, 3 & 4 (W & MHI)		X
	Takahama 1, 2, 3 & 4 (W & MHI)		X
Korea Hydro & Nuclear Power Corp	Kori 1, 2, 3 & 4 (W)		X
	Hanbit 1 & 2 (W)		X
	Hanbit 3, 4, 5 & 6 (CE)		X
	Hanul 3, 4 , 5 & 6 (CE)		X
Kyushu	Genkai 2, 3 & 4 (MHI)		X
	Sendai 1 & 2 (MHI)		X
Nuklearna Electrarna KRSKO	Krsko (W)		X
Ringhals AB	Ringhals 2, 3 & 4 (W)		X
Shikoku	Ikata 1, 2 & 3 (MHI)		X
Taiwan Power Co.	Maanshan 1 & 2 (W)		X

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TABLE OF CONTENTS

List of Tables

List of Figures

List of Acronyms

B&W	Babcock and Wilcox
CCDP	conditional core damage probability
CCL	critical crack length
CCNPP	Calvert Cliffs Nuclear Power Plant
CDF	core damage frequency
CE	Combustion Engineering
DEGB	double ended guillotine break
DLE	design limiting events
FCG	fatigue crack growth
FSAR	final safety analysis report
FSAR	final safety analysis report
GQA	graded quality assurance
ISI	inservice inspection
IST	inservice testing
LBB	leak-before-break
LERF	large early release frequency
LOCA	loss of coolant accident
LOOP	loss offsite power
MT	magnetic particle testing
NDE	non-destructive examination
NRC	Nuclear Regulatory Commission
OD	outer diameter
PMSC	Pump & Motor Services
PRA	probabilistic risk assessment
PROF	probability of failure
PT	penetrant testing
PWROG	Pressurized Water Reactor Owners Group
RCP	reactor coolant pump
RCPM	reactor coolant pump motor
RCS	reactor coolant system
RG	regulatory guide
rpm	revolutions per minute
RT_{NDT}	reference nil-ductility transition temperature
SER	safety evaluation report
SLR	subsequent license renewal
SRP	standard review plan
SRRA	structural reliability and risk assessment
SSCs	systems, structures and components
USAR	updated safety analysis report
UT	ultrasonic examination/ultrasonic test
W	Westinghouse
WOG	Westinghouse Owners Group
W-PROF	Westinghouse PROF Software Library

Record of Revisions

Report	Revision	Date	Revision Description
PWROG-17011-NP	0	Nov. 2017	Original issue.
	1	May 2018	Rev. 1 removes unnecessary contents that are duplicates in WCAP-14535A [1] and WCAP-15666-A [2]. All evaluation results and conclusions are unchanged.
	2	Jan. 2019	Rev. 2 addresses NRC Request for Additional Information (RAI). In seeking to address the NRC's RAI, it was determined that the RPFWPROF executable file used in PWROG-17011-NP, Revisions 0 and 1 cannot reproduce the results of WCAP-15666-A when run on original computer platforms. Westinghouse reestablished configuration and control, verified and validated the RPFWPROF program. The risk assessment is revised in Section 3 based on the corrected RPFWPROF runs. The deterministic evaluations in Section 2 are unchanged. The error is captured in Westinghouse CAP system. Specific revised pages are p.3-9 to 3-11, 3-13 to 3-17; Figures 3-4 to 3-6; p. 3-22 to 3-23, 3-25 to 3-32, and 5-1.
PWROG-17011-NP-A	2	Oct. 2019	The content of PWROG-17011-NP-A, Rev. 2 is the same as PWROG-17011-NP, Rev. 2, with the addition of the U.S. NRC safety evaluation report in the beginning of this report, and RAI responses in Appendix B.

1 INTRODUCTION

The purpose of this topical report (TR) is to extend the applicability of WCAP-14535A [1] and WCAP-15666-A [2] to subsequent license renewal (SLR), i.e., 80 years of operation.

Westinghouse provided the technical basis in WCAP-14535A [1] for the elimination of inspection requirements for the reactor coolant pump (RCP) motor flywheels for all operating domestic Westinghouse and several B&W plants. The NRC issued a Safety Evaluation Report (SER) in September 12, 1996, accepting the technical arguments but did not allow for total elimination of examinations as WCAP-14535A [1] requested. The SER provided partial relief from the reactor coolant pump (RCP) motor flywheels examination requirements in NRC RG 1.14 [3], by allowing an extension in the examination frequency from 40 months to 10 years It further relaxed the RG 1.14 examination guidance by recommending an in-place ultrasonic examination (UT) over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or an alternative surface examination, i.e., magnetic particle testing (MT) and/or liquid penetrant testing (PT), of the exposed surfaces defined by the volume of disassembled flywheel. As Section 3.6 of the SER for [1] stated, NRC staff relied solely on the deterministic methodology to review the submittal. The risk assessment was not included in [1] and was not reviewed. WCAP-14535A [1] is applicable to the RCP motor flywheels in all domestic Westinghouse nuclear steam supply system (NSSS) plants, and Oconee Units 1, 2, and 3, Davis Besse, and Three Mile Island Unit 1, which are Babcock and Wilcox (B&W) NSSS plants.

WCAP-15666-A [2] is a follow-up TR that justified extending the 10-year inspection frequency that was approved by the NRC in WCAP-14535A [1] to 20 years. WCAP-15666-A [2] demonstrated that the deterministic results in WCAP-14535A [1] remain valid, and also performed a failure probability analysis to show that the change in risk for a 20-year inspection frequency meet the RG 1.174 [5] acceptance guidelines. The NRC SER for WCAP-15666-A [2] concluded that both the deterministic and probabilistic calculations contained in [2] were acceptable, and approved the 20-year inspection frequency.

WCAP-15666-A [2] is applicable to plants with Westinghouse-designed NSSS plants. Although it included some data for B&W NSSS plants, however, the TR and the NRC SER did not specifically address the applicability of the risk assessments and other evaluations to the three B&W NSSS plants that WCAP-14535-A [1] was applicable to. The following is a quote from the NRC SER for [2].

> *"The NRC staff acknowledges that some of the supporting material for TSTF-421 may also help to support plant-specific applications for the B&W units included in portions of WCAP-15666. The NRC staff will work with licensees for the applicable B&W units to ensure that our processes work as efficiently as possible for those applying for license amendments similar to that described in TSTF-421. The affected licensees are encouraged to discuss this matter with the NRC staff before submitting an application."*

This same applicability is carried over for the TR presented herein. This TR is not applicable to Combustion Engineering (CE) NSSS plants, with the exception of Calvert Cliffs Units 1 and 2. This TR is applicable to Calvert Cliff Units 1 and 2 as these plants have Westinghouse RCP motors and flywheels However, these flywheels and motor operating speeds are different than those evaluated in WCAP-15666-A [2]. Westinghouse performed a plant-specific evaluation for Calvert Cliffs Unit 1 and 2, that applied the using the same methods detailed in WCAP-15666-A [2] for 60 years of operation. This 60-year evaluation is extended to 80 years of operation in this TR.

Revision 1 of this TR removes unnecessary contents that are duplicates in WCAP-14535A [1] and WCAP-15666-A [2]. Change bars are not used. All evaluation results and conclusions are unchanged.

Revision 2 of this TR addresses NRC Request for Additional Information on Turkey Point Subsequent License Renewal (Set 5, RAI 4.3.5-2). In seeking to address the NRC's RAI, Westinghouse engineers determined that the RPFWPROF executable file used in PWROG-17011-NP cannot reproduce the results of WCAP-15666-A when run on original computer platforms.

To correct for this, Westinghouse has performed a review of the deterministic aspects of the analysis to ensure their continued appropriateness, re-establish configuration control of the RPFWPROF program and determined revised RCP Flywheel probabilities of failure for 40, 60, and 80 years of operation. In efforts to address NRC's question regarding the basis for the K_{IC} model in the RPFWPROF probabilistic assessment, an error was uncovered in the available hard copy of the original independently reviewed source code for RPFWPROF. Removing computer platform differences, the RPFWPROF results in WCAP-15666-A were reproduced, then the error in the median K_{IC} model was corrected per [15] This error is captured in the Westinghouse corrective action program (CAP). The corrected RPFWPROF was verified and validated per Westinghouse software control procedures.

Revision 2 of this TR revised the risk assessment in Section 3 based on the corrected RPFWPROF runs. The deterministic evaluations in Section 2 are unchanged.

2 BACKGROUND

2.1 DESIGN AND FABRICATION

Westinghouse RCP motor flywheels consist of two large steel discs that are shrunk fit directly to the RCP motor shaft. The individual flywheel discs are bolted together to form an integral flywheel assembly, which is located above the RCP rotor core. Typically, each flywheel disc is keyed to the motor shaft by means of three vertical keyways, positioned at 120° intervals. The bottom disc usually has a circumferential notch along the outside diameter bottom surface for placement of anti-rotation pawls. See Figure 2-1 for the configuration of a typical Westinghouse flywheel.

Westinghouse has manufactured the RCP motors for all operating Westinghouse plants. All of the RCP motor flywheels for Westinghouse plants are made of SA-533 Grade B Class 1 steel. As in WCAP-15666-A [2], a range of RT_{NDT} values from 0°F to 60°F was assumed in the integrity evaluations of [1], which are discussed later in this report.

Westinghouse designed flywheels are also used for Calvert Cliffs Units 1 and 2. They will be addressed separately in Section 3 for the risk assessment, and in Appendix A for the deterministic evaluations.

Consistent with the evaluations performed in [1], larger flywheel outside diameter for the flywheel assembly is used in this TR, because it is conservative with respect to stress and fracture.

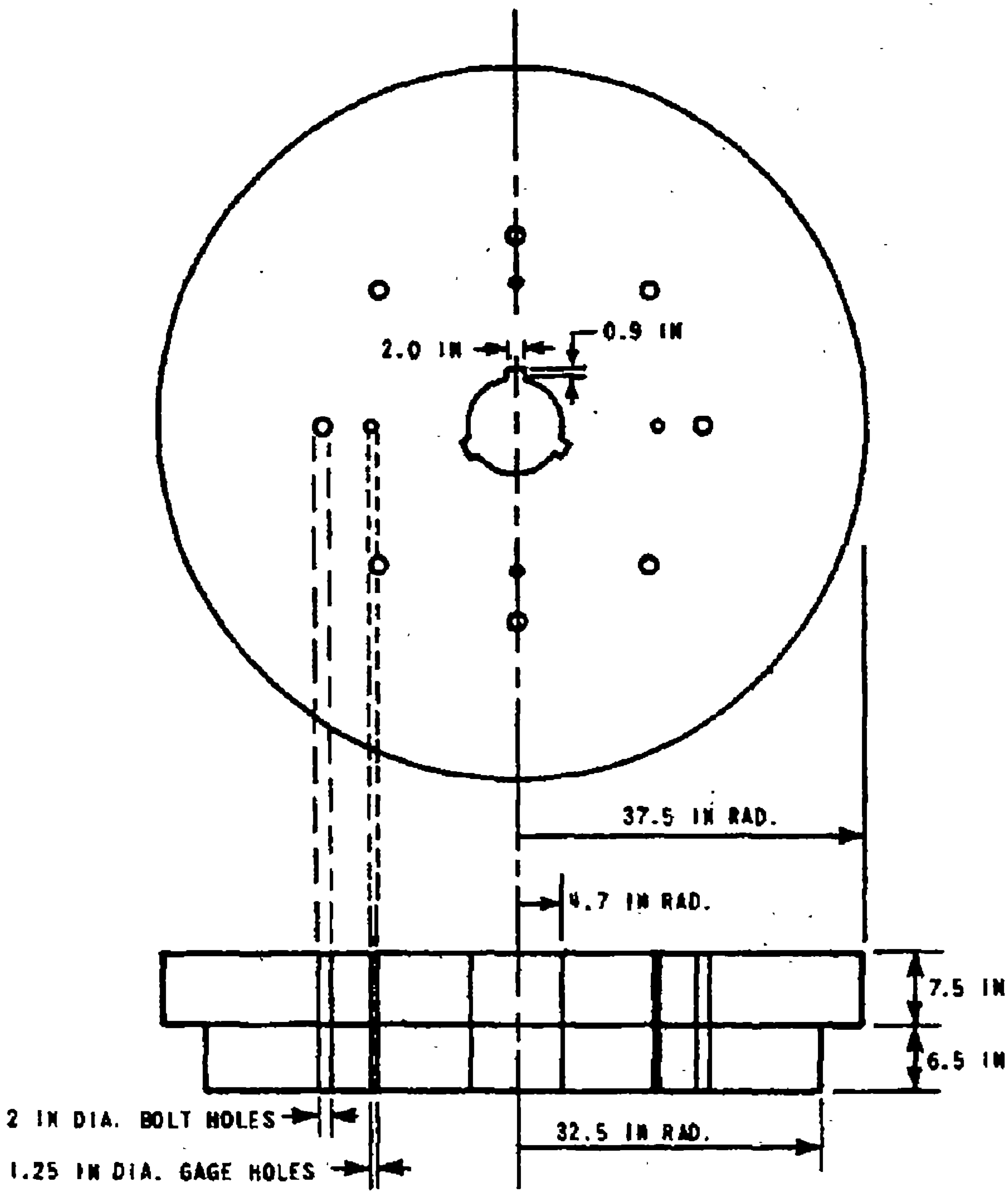

Figure 2-1: Example of a Typical Westinghouse RCP Motor Flywheel

2.2 INSPECTION

Flywheels are inspected at the plant or during motor refurbishment at an offsite facility. Inspections are conducted under the ASME Boiler and Pressure Vessel Code, Section XI [4], which identifies the standard practice for control of instrumentation and personnel qualification. Ultrasonic test (UT) level II and III examiners conduct the inspections.

WCAP-15666-A [2] discussed the examination volume, approach, access and exposure in detail. This discussion remains applicable for SLR.

Inspection History

The flywheel inspection results and the summary of recordable indications from the MUHP-5042 study are presented in Table 2-3 and Table 2-4 of WCAP-15666-A [2]

Inspection History Update

A summary of all Westinghouse RCP flywheels that were inspected by Framatome (formerly AREVA) is summarized in Table 2-1.

Four RCP flywheels where determined to have recordable indications. All four indications were determined to be non-relevant; no repairs were required to be performed on any of those RCP flywheels.

The four recordable indications are discussed in Table 2-2.

Table 2-1: RCP Flywheel Inspection Data

Plant	Number of Flywheels	Total Number of Flywheel Inspections	Total Number of Inspection with No Indications or Non-recordable Indications	Total Number of Inspection With Recordable Indications	Number of Indications Affecting Flywheel Integrity
A	9	9	8	1	0
B	9	9	8	1	0
C	2	2	2	0	0
D	9	14	14	0	0
E	2	2	2	0	0
F	3	3	3	0	0
G	7	7	7	0	0
H	13	13	13	0	0
I	1	1	1	0	0
J	9	9	8	1	0
K	1	1	1	0	0
L	8	9	8	1	0
M	1	1	1	0	0
N	1	1	1	0	0
Total	**75**	**81**	**77**	**4**	**0**

Table 2-2: Flywheel Inspection Data Recordable Indications

Plant	Year	Description of Recordable Indications
A	2015	A Recordable UT indication – Accepted. Lamination with 50% of back wall loss 1" x 4".
B	2006	Procedurally recordable UT indications were identified in the bottom flywheel plate during the 45 degree shear wave examination – Accepted per NB-2530.
J	2005	Indications were identified in two of three keyways in the lower thickness. The indications were dispositioned as acceptable because they are considered to be "non-relevant due to the machining process."
L	2012	These were determined to be non-relevant indications. There were several low amplitude responses that were identified during the radial examinations. These responses were indicative of small machine grooves or marks that extend 360° around the flywheel.

2.3 STRESS AND FRACTURE EVALUATION

Section 2.3 of WCAP-15666-A [2] summarized the stress and fracture evaluation. The ductile and brittle failure mechanisms were considered in flywheel evaluation. The methodology is unchanged for this TR. The evaluation requirements are per RG 1.14 [3].

2.3.1 Selection of Flywheel Groups for Evaluation

As discussed in [2], stresses in the flywheel are a strong function of the outer diameter (approximately proportional to the square of the OD dimension). Therefore, the two groups shown in Table 2-3 with the largest flywheel outer diameter (Groups 1 and 2) bound all other groups defined in WCAP-15666-A [2], and were selected for the deterministic and probabilistic evaluations.

Table 2-3: Flywheel Groups Evaluated for Program MUHP-5043 [2]

Flywheel Evaluation Group	Outer Diameter (inch)	Bore (inch)	Keyway Radial Length (inch)	Comments
1	76 50	9.375	0.937	Maximum OD
2	75.75	8.375	0.906	Large OD, minimum bore.

2.3.2 Ductile Failure Analysis

The flywheel stresses are dependent on dimensions and rotation speed. Extending the operating period to 80 years does not affect the stress calculation. Therefore, the ductile failure analysis in [2] remains valid for 80 years of operation.

These results from [2] are summarized in Table 2-4. The RG 1.14 acceptance criteria for ductile failure of the flywheels are satisfied.

Table 2-4: Ductile Failure Limiting Speed

Flywheel Evaluation Group	Assuming No Cracks		Crack Length (as measured from the maximum radial location of the keyway)			
	Neglecting Keyway Radial Length	Considering Keyway Radial Length	1" Crack	2" Crack	5" Crack	10" Crack
1	3487	3430	3378	3333	3240	3012
2	3553	3493	3435	3386	3281	3060

2.3.3 Non-ductile Failure Analysis

The flywheel stress intensity factor, K_I, is dependent on geometry, postulated flaw dimensions and stress condition (due to rotation speed). Extending the operating period to 80 years does not affect the K_I calculations. Furthermore, the flywheel is not local or adjacent to the reactor core; therefore, the effect of irradiated embrittlement is negligible, and the fracture toughness, K_{Ic} does not change due to the 80-year extension. Therefore, the non-ductile failure analysis in [2] remains valid for 80 years of operation.

The results from [2] are shown in Table 2-5. The ambient temperature of 70°F was conservatively used as the operating temperature, while the typical containment ambient temperature is 100°F to 120°F. At the maximum flywheel overspeed condition of 1500 rpm (considering LBB), the critical crack lengths were calculated for cracks emanating radially from the keyway. The crack length is defined as radially from the keyway. The percentage through the flywheel is defined as the crack length divided by the radial length from the maximum radial keyway location to the flywheel outer radius, i.e., percentage through-wall. The critical crack lengths are quite large, even when considering higher values of RT_{NDT} and a lower than expected operating temperature.

Table 2-5: Critical Crack Lengths for Flywheel Overspeed of 1500 rpm (Considering LBB)

Flywheel Evaluation Group	Critical Crack Length in Inches and % through Flywheel		
	RT_{NDT} = 0°F	RT_{NDT} = 30°F	RT_{NDT} = 60°F
1	16.6" (50%)	7.7" (24%)	3.1" (9%)
2	17.5" (53%)	8.5" (26%)	3.6" (11%)

2.3.4 Fatigue Crack Growth

FCG is dependent on the flywheel K_I at operating and rest states (ΔK_I), and the number of start and shutdown cycles. As discussed previously, the 80-year extension has no impact on the K_I calculations. The 6000 cycles used in the FCG calculation of [2] was determined to be conservative for 80 years of operation because it is unlikely the RCP would go through a more than 6 start and stop cycles every month for 80 years. However, the 6000 cycles for 80 years of operation must be confirmed to be applicable on a plant-specific basis.

The FCG calculations assumed the 6000 cycles of RCP start and shutdown for the 80-year plant life. The FCG results from [2] are applicable and are shown in Table 2-6 The crack growth is negligible over an 80-year life of the flywheel, even when assuming a conservative initial crack length as shown in Table 2-6.

Table 2-6: Fatigue Crack Growth Assuming 6000 RCP Starts and Stops

Flywheel Evaluation Group	Flywheel OD (inch)	Flywheel Bore (inch)	Keyway Radial Length (inch)	Length From Keyway to OD (inch)	Assumed Initial Crack Length (inch)	ΔK_I (ksi$\sqrt{in}$)	Crack Growth after 6000 cycles (inch)
1	76.50	9.375	0.937	32 63	3.26	38	0.08
2	75.75	8.375	0 906	32.78	3 28	37	0.08

2.3.5 Excessive Deformation Analysis

The deformation of the flywheel is only dependent on the rotation speed and physical attributes of the flywheel. The 80-year extension has no impact on the excessive deformation analysis of the flywheel. The results in [2] remain applicable to 80 years of operation.

At the flywheel over speed condition of 1500 rpm (157.08 radians/second), the change in the bore radius and outer radius is shown in Table 2-7. A maximum deformation of 0.006 inch is anticipated for the flywheel over speed condition. As deformation is proportional to the square of angular speed, ω^2, this represents an increase of 56% over the normal operating deformation of 0.004 inch. This increase would not result in any adverse conditions such as excessive vibrational stress leading to crack propagation, since the flywheel assemblies are typically shrunk fit to the flywheel shaft, and the deformations calculated are negligible

Table 2-7: Flywheel Deformation at 1500 rpm

Flywheel Evaluation Group	Change in Bore Radius (inch)	Change in Outer Radius (inch)
1	0.003	0.006
2	0 003	0.006

2.4 SUMMARY OF STRESS AND FRACTURE RESULTS

The deterministic integrity evaluations in WCAP-15666-A [2] remain applicable for 80 years of operation. The evaluations concluded that the RCP motor flywheels have a very high tolerance for the presence of flaws, especially with the 1500 rpm overspeed due to the application of LBB [2]. As noted in [2], the probabilistic assessment evaluates all credible flywheel speeds. This TR uses the same probabilistic assessment methodology as [2], which is discussed in Section 3.

There are no significant mechanisms for inservice degradation of the flywheels, since they are isolated from the primary coolant environment. The evaluations presented in this section have shown there is no significant deformation of the flywheels, even at maximum overspeed conditions. FCG calculations have shown that even with a large assumed flaw, the crack growth for 80 years of operation is negligible. Therefore, based on these deterministic evaluations, the flywheel inspections completed following manufacture and prior to service are sufficient to ensure their integrity during 80 years of service. As discussed in Section 2.2 and [1 and 2], the most likely source of inservice degradation is damage to the keyway region that could occur during disassembly or reassembly for refurbishment and inspection.

3 RISK ASSESSMENT

The quantitative risk assessment discussed below provides the justification for applying the WCAP-15666-A [2] 20-year flywheel inspection interval for 80 years of operation. Specifically, the risk analyses confirms that applying the inspection extension to flywheels in operation up to 80 years has a negligible impact on risk (CDF and LERF), i.e , it is within the risk acceptance criteria of RG 1.174 [5]. This section provides a discussion on the requirements of [5], and extends the previous flywheel failure probability assessment in [2] to 80 years of operation.

3.1 RISK-INFORMED REGULATORY GUIDE 1.174 METHODOLOGY

The NRC risk-informed regulatory framework for modifying a plant's licensing basis is contained in RG 1.174, Revision 2 [5]. The intent of this risk-informed process is to allow insights derived from probabilistic risk assessments to be used in combination with traditional engineering analysis to focus licensee and regulatory attention on issues commensurate with their importance to safety. Additional regulatory guidance is contained in [6].

The approach described in RG-1.174 is used in each of the application-specific RGs/SRPs, and has four basic steps as shown in Figure 3-1. The four (4) basic steps are discussed below.

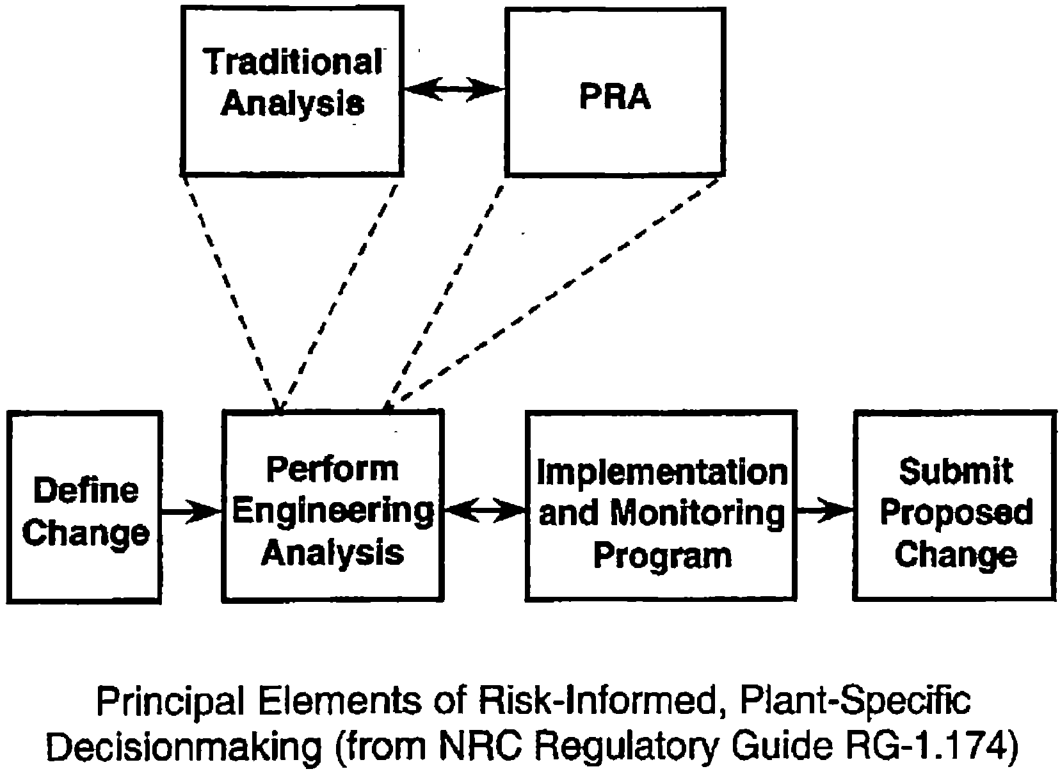

Figure 3-1: NRC Regulatory Guide 1.174 Basic Steps

Step 1: Define the proposed change

This element includes identifying:

1. Those aspects of the plant's licensing bases that may be affected by the change
2. All systems, structures, and components (SSCs), procedures, and activities that are covered by the change and consider the original reasons for inclusion of each program requirement
3. Any engineering studies, methods, codes, applicable plant-specific and industry data and operational experience, PRA findings, and research and analysis results relevant to the proposed change.

Step 2: Perform engineering analysis

This element includes performing the evaluation to show that the fundamental safety principles on which the plant design was based are not compromised (defense-in-depth attributes are maintained) and that sufficient safety margins are maintained. The engineering analysis includes both traditional deterministic analysis and probabilistic risk assessment. The evaluation of risk impact should also assess the expected change in CDF and LERF, including a treatment of uncertainties. The results from the traditional

analysis and the probabilistic risk assessment must be considered in an integrated manner when making a decision.

Step 3: Define implementation and monitoring program

This element's goal is to assess SSC performance under the proposed change by establishing performance monitoring strategies to confirm assumptions and analyses that were conducted to justify the change.

This is to ensure that no unexpected adverse safety degradation occurs because of the changes. Decisions concerning implementation of changes should be made in light of the uncertainty associated with the results of the evaluation. A monitoring program should have measurable parameters, objective criteria, and parameters that provide an early indication of problems before becoming a safety concern. In addition, the monitoring program should include a cause determination and corrective action plan.

Step 4: Submit proposed change

This element includes:

1. Carefully reviewing the proposed change in order to determine the appropriate form of the change request
2. Assuring that information required by the relevant regulation(s) in support of the request is developed
3. Preparing and submitting the request in accordance with relevant procedural requirements.

Five (5) fundamental safety principles are described which should be met for each application for a modification. These are shown in Figure 3-2 and are discussed below.

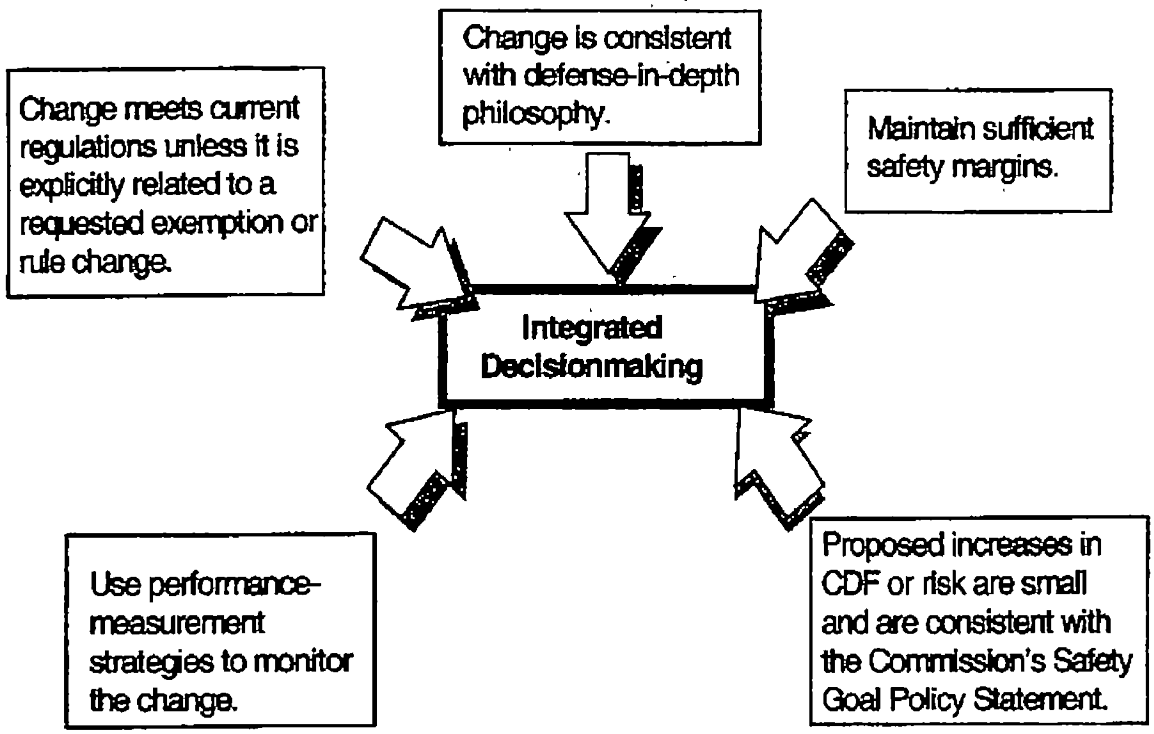

Figure 3-2: Principles of Risk-Informed Regulation [5]

Principle 1: Change meets current regulations unless it is explicitly related to a requested exemption or rule change

The proposed change is evaluated against the current regulations (including the general design criteria) to either identify where changes are proposed to the current regulations (e.g., technical specification, license conditions, and FSAR), or where additional information may be required to meet the current regulations.

Principle 2: Change is consistent with defense-in-depth philosophy

Defense-in-depth has traditionally been applied in reactor design and operation to provide a multiple means to accomplish safety functions and prevent the release of radioactive material. As defined in RG-1.174, defense-in-depth is maintained by assuring that:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences to the system (e.g., no risk outliers)
- Defenses against potential common cause failures are preserved and the potential for introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded (the barriers are identified as the fuel cladding, reactor coolant pressure boundary, and containment structure)
- Defenses against human errors are preserved

Defense-in-depth philosophy is not expected to change unless:

- A significant increase in the existing challenges to the integrity of the barriers occurs
- The probability of failure of each barrier changes significantly,
- New or additional failure dependencies are introduced that increase the likelihood of failure compared to the existing conditions, or
- The overall redundancy and diversity in the barriers changes.

Principle 3: Maintain sufficient safety margins

Safety margins must also be maintained. As described in RG-1.174, sufficient safety margins are maintained by assuring that:

- Codes and standards, or alternatives proposed for use by the NRC, are met, and
- Safety analysis acceptance criteria in the licensing basis (e.g , FSARs, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

Principle 4: Proposed increases in CDF or risk are small and are consistent with the Commissions Safety Goal Policy Statement

To evaluate the proposed change with regard to a possible increase in risk, the risk assessment should be of sufficient quality to evaluate the change. The expected change in CDF and LERF are evaluated to address this principle. An assessment of the uncertainties associated with the evaluation is conducted. Additional qualitative assessments are also performed.

There are two acceptance guidelines, one for CDF and one for LERF, both of which should be used.

The guidelines for CDF are:

- If the application can be clearly shown to result in a decrease in CDF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to CDF.
- When the calculated increase in CDF is very small, which is taken as less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF.

- When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year.
- Applications which result in increases to CDF above 10^{-5} per reactor year would not normally be considered.

AND

The guidelines for LERF are:

- If the application can be clearly shown to result in a decrease in LERF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to LERF
- When the calculated increase in LERF is very small, which is taken as being less than 10^{-7} per reactor year, the change will be considered regardless of whether there is a calculation of the total LERF.
- When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year.
- Applications which result in increases to LERF above 10^{-6} per reactor year would not normally be considered.

These guidelines are intended to provide assurance that proposed increases in CDF and LERF are small and are consistent with the intent of the Commission's Safety Goal Policy Statement.

Principle 5: The impact of the proposed change should be monitored using performance-measurement strategies to monitor the change

Performance-based implementation and monitoring strategies are also addressed as part of the key elements of the evaluation as described previously.

The following sections address the principle elements of the RG-1.174 process and the principles of risk-informed regulation to RCP motor flywheel examination frequency reduction.

3.2 FAILURE MODES AND EFFECTS ANALYSIS

A failure modes and effects analysis is used to identify the potential failure modes of a RCP motor flywheel and the effect that each failure mode would have on the plant SSCs in relation to overall plant safety.

Failure Modes

The primary failure mode of the RCP motor flywheel is growth of an undetected fabrication induced flaw in the keyway of the flywheel that emanates radially from that location to a point such that it reaches a critical flaw size during normal or accident conditions. Once the critical flaw size is reached during plant operation, the flywheel has the potential to catastrophically fail, resulting in flywheel fragments, which are essentially high energy missiles that could impact other SSCs important to plant safety. The growth of a flaw is primarily related to stresses generated from changes in the flywheel speed. The flywheel inspection process, which itself has the potential to introduce flywheel damage as discussed in [1], is not considered in the assessment. This is because the purpose of the assessment is to support interval extension, which will reduce unnecessary occurrences for introducing potential damage.

As discussed in [1], the normal operating speed of the RCP motor flywheel for Westinghouse RCPs is 1189 revolutions per minute (rpm), with a synchronous speed of 1200 rpm. It is designed for an overspeed of 1500 rpm, which is 125% of the synchronous speed. The flywheel speed can also vary as a result of plant events, including accidents such as a double ended guillotine break (DEGB) in the main reactor coolant loop piping.

Westinghouse designed flywheels are also used for Calvert Cliffs Units 1 and 2. These plants include Byron-Jackson designed pumps and motors and therefore have different normal flywheel operating speeds and different flywheel accident responses. The normal operating speed of the RCP motor flywheel for these RCPs is 900 rpm, with a design limiting speed of 1125 rpm. The maximum overspeed following a design basis LOCA is limited to 1368 rpm as stated in the Calvert Cliffs UFSAR.

When operating as a motor, the rotor of a polyphase induction machine rotates in the direction of, but slightly lower than, the rotating magnetic flux provided by the stator. This slight speed difference is typically expressed in percent and designated slip. If the shaft of the machine is driven above synchronous speed by a prime mover (with line voltage maintained on the stator) the rotor conductors rotate faster than the magnetic flux and the slip becomes negative. The rotor current and consequently the stator current reverse under the condition of negative slip and the machine operates as an induction, or asynchronous, generator. The RCP motor functions as an efficient torque producer under normal conditions. In the unlikely event that a hydraulic torque is applied to the motor shaft in the direction of increasing shaft speed (thus acting as a prime mover), the slip would become negative and, with the stator connected to the grid, the motor would function as a dynamic brake.

If the power supply to the motor is interrupted (zero voltage), the motor torque would be reduced to a negligible value, since torque is proportional to the supplied voltage.

However, a design feature of Westinghouse NSSS plants ensures that the electrical power supply to the RCP will be maintained for at least 30 seconds after a turbine trip following a LOCA. This design feature is also maintained following a loss of offsite power (LOOP); for the expected case of available off-site power, power to the RCP would continue through the LOCA transient. As a result, reverse torque is provided.

Westinghouse also performed several sensitivity studies to evaluate the effect of the break opening area on the RCP flywheel speed for typical Westinghouse NSSS plants. Specifically, break sizes equal to a DEGB of the main coolant piping, 60% of that DEGB, and a 3 ft^2 have been analyzed. A 3 ft^2 break size corresponds to a pipe of approximately 23 inches in inside diameter; the only RCS piping greater than this, is the main coolant loop piping. The first two breaks have blowdown times equal to or less than the RCP trip time; therefore, the applied voltage prevents overspeed. The latter break has an extended blowdown time, but the RCP flow at the time of RCP trip is reduced such that the speed decreases. Smaller breaks are not limiting even though the voltage is maintained for only 30 seconds. Results of these studies were discussed in [1].

To investigate the consequences of RCP overspeed, [2] analyzed a spectrum of LOCA events resulting in a range of flywheel transients. Results of that analysis indicated that the limiting event was the DEGB with an instantaneous loss of power, this led to a peak flywheel speed of 3321 rpm. It was also noted that the 3 ft^2 break area case showed a decrease in speed such that the normal operating speed is not exceeded.

Based on the WCAP-15666-A assessments, the following scenarios are associated with the primary mode of potential failure in the Westinghouse RCP motor flywheel that are related to operating speed and potential overspeed during various conditions:

- Failure during normal plant operation resulting in a plant trip (1200 rpm peak speed)
- Failure of the RCP motor flywheel associated with a plant transient or LOCA event with no loss of electrical power to the RCP (1200 rpm peak speed)
- Failure of the RCP motor flywheel associated with a plant transient or LOCA event (up to 3 ft^2 with an instantaneous loss of electrical power to the RCP (1200 rpm peak speed)
- Failure of the RCP motor flywheel associated with a DEGB coincident with an instantaneous loss of electrical power, such as LOOP (3321 rpm peak speed). This case bounds and is conservatively applied to all flywheel transients for LOCA break areas.

WCAP-15666-A [2] was limited in scope to RCPs with Westinghouse supplied pumps and flywheels. It is also the intent of this topical report to extend the applicability of the flywheel inspection extension to Calvert Cliffs Units 1 and 2 which contain Byron Jackson RCPs but use Westinghouse supplied flywheels. It is important to note that as a result of significant design differences between the Westinghouse and Calvert Cliffs units, the Calvert Cliffs RCP operational and transient conditions are different. Specifically, Calvert Cliffs pumps normally operate at 900 rpm with a design speed of

1125 rpm. Furthermore, the peak RCP post LOCA speed is limited to 1368 rpm. Therefore, the Calvert Cliffs Units 1 and 2 Pump/Flywheel Combinations analyses were based on the following:

- Failure during normal plant operation resulting in a plant trip (1125 rpm peak speed)
- Failure of the RCP motor flywheel associated with a plant transient or LOCA event with no loss of electrical power to the RCP (1125 rpm peak speed)
- Failure of the RCP motor flywheel associated with a plant transient or LOCA event (up to 3 ft^2) with an instantaneous loss of electrical power to the RCP (1125 rpm peak speed)
- Failure of the RCP motor flywheel associated with a DEGB coincident with an instantaneous loss of electrical power, such as, loss of offsite power (LOOP) (1368 rpm peak speed). As for the Westinghouse flywheel analysis, this case is conservatively assumed to bound all flywheel transients for LOCA break areas resulting from equivalent reactor coolant pipe breaks greater than a 3.0 ft^2 break and less than a double ended break.

Failure Effects

The failure of the RCP motor flywheel during normal plant operation would directly result in a reactor trip. However, the potential indirect or spatial effects associated with a postulated flywheel failure present a greater challenge in terms of failure effects or consequences. As discussed previously, the flywheel has the potential to catastrophically fail, resulting in flywheel fragments, which are essentially high energy missiles, which could impact other SSCs important to plant safety. Failure of these other SSCs could potentially impact the overall plant safety in terms of core damage (e.g., as a result of the loss of safety injection) or large, early release (as a result of potential impacts on containment structures or systems).

In order to address plant specific design differences on a generic basis, it is conservatively assumed that failure of the RCP motor flywheel results in core damage and a large early release, i.e., the flywheel failure frequency is equal to CDF and LERF.

Section 3.3 discusses the process for estimating the likelihood of the primary failure mode of the RCP motor flywheel. Section 3.4 then combines this failure probability estimation with the likelihood of various plant events and consequences to estimate the change in risk for extending the flywheel examination interval from 10 years to 20 years, for RCP/Flywheels in service up to 80 years.

3.3 FLYWHEEL FAILURE PROBABILITY

The quantitative risk assessment discussed below updates the risk assessment performed in WCAP-15666-A [2] and provides the justification for extending the 20-year flywheel inspection interval for 80 years of operation. Specifically, the risk analyses confirms that applying the inspection extension to flywheels in operation up to 80 years

has a negligible impact on risk (CDF and LERF), i.e , it is within the risk acceptance criteria of RG 1.174 [5]. The update of the WCAP-15666 analysis was necessary as it was discovered that the equation used to define the flywheel fracture toughness (K_{IC}) was incorrectly programmed in the probabilistic fracture mechanics (PFM) code RPFWPROF used to establish the probability of flywheel failure. This section provides a brief description of the code change and applies the revised code to demonstrate that the conclusions from WCAP-15666-A remain valid and that the 20 year flywheel inspection interval can be extended to 80 years of operation. A discussion of the change to the PFM model is discussed in Section 3.3.1

The risk assessments in this section apply to all Westinghouse RCP/flywheels, as well as the Calvert Cliffs Units 1 and 2 RCP/flywheels which contain a Byron Jackson [13] RCP with a Westinghouse flywheel.

To investigate the effect of flywheel inspections on the risk of failure, a structural reliability and risk assessment is performed for flywheels with up to 80 years of operation. Twelve (12) month operating (or fuel) cycles are conservatively assumed for the evaluation. This section discusses the methodology used and summarizes the results from this assessment.

As described in Section 3.2, the Westinghouse RCP has a normal operating speed of 1189 rpm, a synchronous speed of 1200 rpm, and an overspeed of 1500 rpm, considering LBB [2]. Therefore, a peak speed of 1500 rpm is conservatively used in the evaluation of RCP motor flywheel integrity to represent all conditions except a DEGB coincident with an instantaneous loss of electrical power. For this more limiting event, a peak speed of 3321 rpm is used.

The structural reliability evaluation for a Westinghouse RCP utilizes the work previously performed and summarized in [1], where the 1500 rpm overspeed speed had been assumed. In addition, this evaluation builds upon the initial analysis discussed in [2].

The structural reliability evaluation for the Calvert Cliffs RCP is based on plant-specific analyses and flywheel failure probabilities which are based on nominal and transient flywheel operation at 1125 rpm and a post design basis LOCA flywheel transient overspeed of 1368 rpm.

3.3.1 Method of Calculation Failure Probabilities

The method for calculating flywheel failure probabilities is based on the method in WCAP-15666-A [2]. While there are no changes to methodology, this evaluation corrects a significant error in the flywheel-specific PFM code, RPFWPROF, used for calculation of flywheel failure probability. The WCAP-15666-A version of RPFWPROF included an embedded error for the median value of K_{IC}. Specifically, the RPFWPROF intended to define the median value of K_{IC} as follows:

$$K_{IC} = 55.1 + 28.8 \exp(0.0214 (T-RT_{NDT}) \text{ for } T-RT_{NDT} > -50°F \text{ [15]}$$

However, due to an undetected programming error the K_{IC} parameter was included in RPFWPROF as:

$$K_{IC} = 55.1 + 28.8 \exp(0.214 (T-RT_{NDT})) \text{ for } T-RT_{NDT} > -50\ ^{o}F$$

The implication was to increase the fracture toughness for $T-RT_{NDT} > 0\ ^{o}F$ and decrease the parameter for $T-RT_{NDT} < 0\ ^{o}F$. Re-evaluation of flywheel failure probability for various reactor scenarios indicated that the net effect was to predict lower failure probabilities for flywheels subjected to normal operation, design limiting transients and LOCAs smaller than 0.3 ft^2, and to increase failure probabilities the very low probability large LOCA scenarios (LOCAs > 0.3 ft^2) with loss of off-site power.

To ensure that the extent of condition was limited to this error, hand calculations and EXCEL based analyses were used to confirm predictions of RPFWPROF for representative sample cases.

The following discussion applies to the updated version of RPFWPROF.

The probability of failure of the RCP motor flywheel as a function of operating time t, Pr(t < t1), is calculated directly for each set of input values using Monte-Carlo simulation with importance sampling. The Monte-Carlo simulation does not force the calculated distribution of time to failure to be of a fixed type (e.g., Weibull, Log-normal or Extreme Value). The actual failure distribution is estimated based upon the distributions of the uncertainties in the key structural reliability model parameters and plant specific input parameters. Importance sampling, as described by Witt [7], is a variance reduction technique to greatly reduce the number of trials required for calculating small failure probabilities. In this technique, random values are selected from the more severe regions to increase the probability of an observable failure occurring. However, when a failure is calculated, the count is corrected to account for the lower probability of simultaneously obtaining all of the more severe random values.

The application of the probability of failure methodology is described based on the Westinghouse RPFWPROF program which is generally described in WCAP-14535A [1] and WCAP-15666-A [2].

The description of the key input parameters and associated data used in the RPFWPROF program is presented in Table 3-1 and Table 3-2. Table 3-1 includes the key parameters needed for failure probability calculation. Its usage in the program is specified as shown in the last column of Table 3-1 and schematically in the flow chart of Figure 3-3. "Initial" conditions do not change with time, "Steady-State" is not needed for RPFWPROF, "Transient" calculates fatigue crack growth and "Failure" checks to see if the accumulated crack length exceeds the critical length. In addition, parameter RPM-DLE is included in the model to address the impact of design limiting events (DLE).

Table 3-1: Variables for RCP Motor Flywheel Failure Probability Model			
No.	Name	Description of Input Variable	Usage Type
1	ORADIUS	Outer Flywheel Radius (inch)	Initial
2	IRADIUS	Inner Flywheel Radius (inch)	Initial
3	PFE-PSI	Probability of Flaw Existing (PFE) after Preservice ISI	Initial
4	ILENGTH	Initial Radial Flaw Length (inch)	Initial
5	CY1-ISI	Operating Cycle for First Inservice Inspection	Inspection
6	DCY-ISI	Operating Cycle between Inservice Inspections	Inspection
7	POD-ISI	Flaw Detection Probability per Inservice Inspection	Inspection
8	DFP-ISI	Fraction PFE Increases per Inservice Inspection	Inspection
9	NOTR/CY	Number of Transients per Operating Cycle	Transient
10	DRPM-TR	Speed Change per Transient (RPM)	Transient
11	RATE-FCG	Fatigue Crack Growth Rate (Inch/Transient)	Transient
12	KEXP-FCG	Fatigue Crack Growth Rate SIF Exponent	Transient
13	RPM-DLE	Speed for Design Limiting Event (RPM)	Failure
14	TEMP-F	Temperature for Design Limiting Event (F)	Failure
15	RT-NDT	Reference Nil Ductility Transition Temperature (F)	Failure
16	F-KIC	Crack Initiation Toughness Factor	Failure
17	DLENGTH	Flywheel Keyway Radial Length (Inch)	Failure

Variables 5 to 8 are available to calculate the effects of an ISI in the RPFWPROF program. The effect of ISI calculated using these equations, which are used in the SRRA model for the effect of ISI, are consistent with those described in the pc-PRAISE Code User's Manual [9]. The parameters needed to describe the selected ISI program are the time of the first inspection, the frequency of subsequent inspections (expressed as the number of fuel or operating cycles between inspections) and the probability of non-detection as a function of crack length. For the RCP motor flywheel, the non-detection probability, which is independent of crack length, is simply one minus a constant value of detection probability, variable 7 (POD-ISI) in Table 3-1. An increase in failure probability due to RCP inspection (chance of incorrect disassembly and reassembly) is included in the ISI model but conservatively not used (variable 8 set to zero) in this evaluation.

The median input values and their uncertainties for each of the parameters of Table 3-1 are shown in Table 3-2. The median is the value at 50% probability (half above and half below this value); it is also the mean (average) value for symmetric distributions, like the normal (bell-shaped curve) distribution.

Uncertainties are based upon expert engineering judgment and previous structural reliability modeling experience. For example, the fracture toughness for initiation as a function of the RT_{NDT} and the uncertainties on these parameters are based upon prior probabilistic fracture mechanics analyses of the reactor pressure vessel (RPV) [10]. Also note that the stress intensity factor calculation for crack growth and failure used the flywheel keyway radial length in addition to the calculated flaw length.

Table 3-2: Input Values for RCP Motor Flywheel Failure Probability Model				
No.	**Name**	**Median**	**Distribution**	**Uncertainty***
1	ORADIUS	Per Flywheel Group	Constant	—
2	IRADIUS	Per Flywheel Group	Constant	—
3	PFE-PSI	1.000E-01	Constant	—
4	ILENGTH	1.000E-01	Log-Normal	2 153E+00
5	CY1-ISI	3 000E+00	Constant	—
6	DCY-ISI	4 000E+00	Constant	—
7	POD-ISI	5.000E-01	Constant	—
8	DFP-ISI	0.000E+00	Constant	—
9	NOTR-CY	1.000E+02	Normal	1.000E+01
10	DRPM-TR	1.200E+03 (W) 9 00E+02 (CCNPP)	Normal	1.200E+02 (W) 9.00E+01 (CCNPP)
11	RATE-FCG	9 950E-11	Log-Normal	1.414E+00
12	KEXP-FCG	3.070E+00	Constant	—
13	RPM-DLE**	1.50E+3, 3.321E+3 (W) 1.125E+3, 1.368E+3 (CCNPP)	Normal	1.50E+2, 3.321E+2 (W) 1 125E+2, 1.368E+2 (CCNPP)
14	TEMP-F***	9.500E+01 (W) 7.0E+01 (CCNPP)	Normal	1.250E+01
15	RT-NDT	3 000E+01	Normal	1.700E+01
16	F-KIC	1.000E+00	Normal	1 000E-01
17	DLENGTH	Per Flywheel Group	Constant	—

* The uncertainty is a normal standard deviation, the range (median to maximum) for uniform distributions or the corresponding factor for logarithmic distributions.

** RPM-DLE is modified in each case to allow for risk analysis of various plant conditions and their associated flywheel speeds for both Westinghouse (W) Plants and Calvert Cliffs (CCNPP) Units 1 and 2. The values used for this variable are discussed in Section 3.3 and results of analyses are summarized in Table 3-3.

DLE-RPM Values used for RPFWPROF Analyses		
Plant Design	Transient Speeds (Also used to bounds Normal Operation) (RPM)	Maximum Flywheel Rotational Speed. Large LOCA (RPM)
Westinghouse Plants	1500	3321
Calvert Cliffs	1125	1368

Group specific input variables used in the probability of failure calculations are summarized below:

Flywheel Group	ORadius (inch)	IRadius (inch)	DLength (inch)
Group1	38.25	4.6875	0.937
Group 2	37.875	4.1875	0.906
Calvert Cliffs	41 00	4 719	0.937

Evaluations were performed to determine the effect on the probability of flywheel failure for continuing the previously approved current inservice inspections in accordance with Reference [2] over the life of the plant through 80 years of operation and for discontinuing the inspections. The evaluation also calculated the effects of the inspections being discontinued after ten years. This calculation bounds the effects of any subsequent inservice inspections at 10- to 20-year intervals.

The probability of failure determined by these evaluations is a conservatively calculated parameter because the evaluation conservatively assumes that the probability of a flaw existing after the preservice inspection is 10%, and that the ISI flaw detection probability is only 50%. In reality, most preservice inspection and ISI flaws would be detected, especially for the larger flaw depths which could result in failure. Therefore, the calculated values are very conservative. (The effects of some important parameters on the calculated probability of failure are discussed later in this section). The most important result of the evaluation is the change in calculated probability of failure from continuing versus discontinuing the ISI after 10 years of plant life.

As shown in Figure 3-4, Figure 3-5 and Figure 3-6 and Table 3-3, the ISI provides a negligible benefit for minimizing the potential of failure of the flywheel. The results of this assessment are summarized as follows for a plant life of 40, 60, and 80 years. Note that results presented in Table 3-3 supersedes equivalent information presented in Reference [2].

Table 3-3: Cumulative Probability of Failure over 40, 60 and 80 Years with and without Inservice Inspection

Flywheel Group	Design Limiting Speed (rpm)	Cumulative Probability of Flywheel Failure with ISI at 4-Year Intervals	Cumulative Probability of Flywheel Failure with ISI at 4-Year Intervals Prior to 10 Years and without ISI after 10 Years			% Increase in Cumulative Failure Probability for Eliminating Inspections		
		Over 80 Years	Over 40 Years	Over 60 Years	Over 80 Years	Over 40 Years	Over 60 Years	Over 80 Years
1	1500	1.99E-08	2.00E-08	2.01E-08	2.02E-08	0.95%	1.16%	1 75%
1	3321	5.88E-02	5.88E-02	5.88E-02	5.88E-02	<0.01%	<0.01%	<0.01%
2	1500	1.26E-08	1.26E-08	1.36E-08	1.37E-08	0.51%	8.35%	9.19%
2	3321	1.66E-02	1.66E-02	1.66E-02	1.66E-02	<0.01%	<0 01%	<0.01%
CCNPP	1125	1.14E-10	1.14E-10	1.14E-10	1 14E-10	0.09%	0.09%	0.09%
CCNPP	1368	1.61E-07	1.62E-07	1.63E-07	1 63E-07	0.47%	1.13%	1.28%

As can be seen in Table 3-3, continuing inspection after 10 years has a very minimal impact on the failure probabilities.

Note that for the Westinghouse Group 1 and 2 flywheels subject to Large LOCAs with a consequential LOOP (limiting speed of 3321 rpm), the flywheel failure probability is primarily dependent on the assumed existence of a large unidentified flaw, conservatively selected flywheel properties and operating conditions and the high post-accident flywheel rotational speed. Thus, the probabilistic models predict the same number of failures from the first through the 80th year of operation.

The post LOCA limiting speed of Calvert Cliffs Units 1 and 2 of 1368 RPM resulted in significantly reduced flywheel failure probabilities, when compared with the Westinghouse RCPs.

3.3.2 Sensitivity Study

A sensitivity study was performed to determine the effect of select flywheel risk assessment parameters on the probability of failure, as done in [2]. Consistent with [2], sensitivity studies were performed on a Westinghouse Group 10 flywheel, as this flywheel is representative of average Westinghouse and Byron Jackson flywheel dimensions and configuration. The intent of the sensitivity studies was to illustrate the impact of relatively significant changes to model input parameters on probability of failure predictions. The specific parameters evaluated in this sensitivity study were the probability of detection and the initial flaw length. The results of this study are summarized in the Table 3-4 and sections 3.3.2.1 and 3.3.2.2.

Table 3-4: Effect of Flywheel Risk Parameter on Failure Probability (Flywheel Group 10)

Description of Flywheel Risk Parameter Varied	Probability of Flywheel Failure after 40 years with ISI	Probability of Flywheel Failure after 40 years without ISI
Base Case (Group 10 of [1])	7.3636E-09	7.3709E-09
Probability of Detection of 10%	7.3684E-09	7.3709E-09
Probability of Detection of 80%	7.3625E-09	7.3709E-09
Initial flaw length of 0.05 inches	2 0487E-09	2 0490E-09
Initial flaw length of 0.20 inches	1 0260E-07	1.0314E-07

The values for the base case were for:

- 10% probability of a flaw existing after preservice inspection
- an initial flaw length of 0.10 inch (1.006 inch with keyway)

- an initial ISI at 3 years of plant life, and subsequent inspections at 4-year intervals
- probability of detection of 50% per ISI (see [1], Table 5-5, flywheel Group 10)

A discussion of the results of the sensitivity studies are summarized below.

3.3.2.1 Sensitivity to Change in Flaw Detection Probability

The flaw detection probability was varied from the base case 50% to 10% and 80%. The failure probability increased less than 0.1% for a decrease in flaw detection probability from 50% to 10%. A similarly small increase in failure probability was noted for an increase in flaw detection probability from 50% to 80%. Therefore, the flaw detection probability, which is a measure of how well the inspections are performed, has essentially no effect on the flywheel failure probability.

3.3.2.2 Sensitivity to Initial Flaw Length

The initial flaw length was varied from the base case value of 0.10 inch to 0.05 inch, and 0.20 inch. The failure probability decreased by more than a factor or 3 for a decrease in initial flaw length from 0.10 inch to 0.05 inch, and the failure probability increased an order of magnitude for an increase in initial flaw length from 0.10 inch to 0.20 inch. Therefore, the initial flaw length does affect the flywheel failure probability, but the failure probability remained small, even for larger initial flaw lengths. Moreover, it is expected that the probability of the larger flaw being missed during preservice inspection is smaller than the assumed 10% based on reviews of pre-service inspection records in [2]

3.3.3 Failure Probability Assessment Conclusions

An evaluation of flywheel structural reliability was performed for each of the flywheel groups selected for evaluation following the process outlined in WCAP-15666-A. Using conservative input values for; preservice flaw existence, initial flaw length, inservice flaw detection capability and RCP start/stop transients, it was shown that flywheel inspections beyond ten years of plant life have no significant benefit relative to the probability of flywheel failure. The reasons are that most flaws that could lead to failure would be detected during the preservice inspection or early in the plant life, and the crack growth is negligible over the plant life. It should be noted that the effect on potential flywheel failure from damage through disassembly and reassembly for inspection has not been evaluated. This is because the purpose of the assessment is to support an inspection interval extension, which will reduce unnecessary occurrences for introducing potential damage.

Sensitivity studies showed that improved flaw detection capability and more inspections result in a small relative change in the calculated failure probability. The failure probability is most affected by the initial flaw length and its uncertainty. These parameters are determined by the accuracy of the preservice inspection. The uncertainty could be reduced using the results from the first inservice inspection, but would probably not change much during subsequent inspections.

The failure probability estimates identified in [2] show that inspections after 10 years have a very minimal impact on the failure probabilities. These results bound the effects of any subsequent ISI at 10 to 20 year intervals. No credit has been taken for other indications of potential degradation such as pump vibration monitoring and pump maintenance.

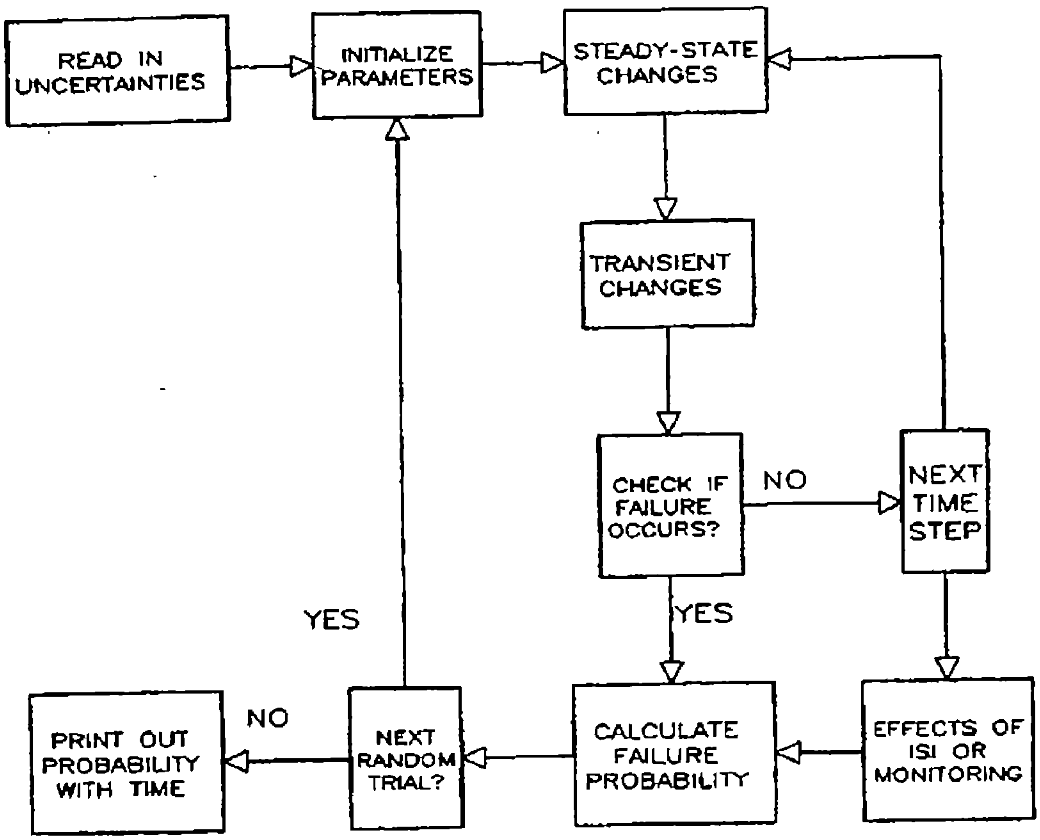

Figure 3-3: Westinghouse PROF Program Flow Chart for Calculating Failure Probability

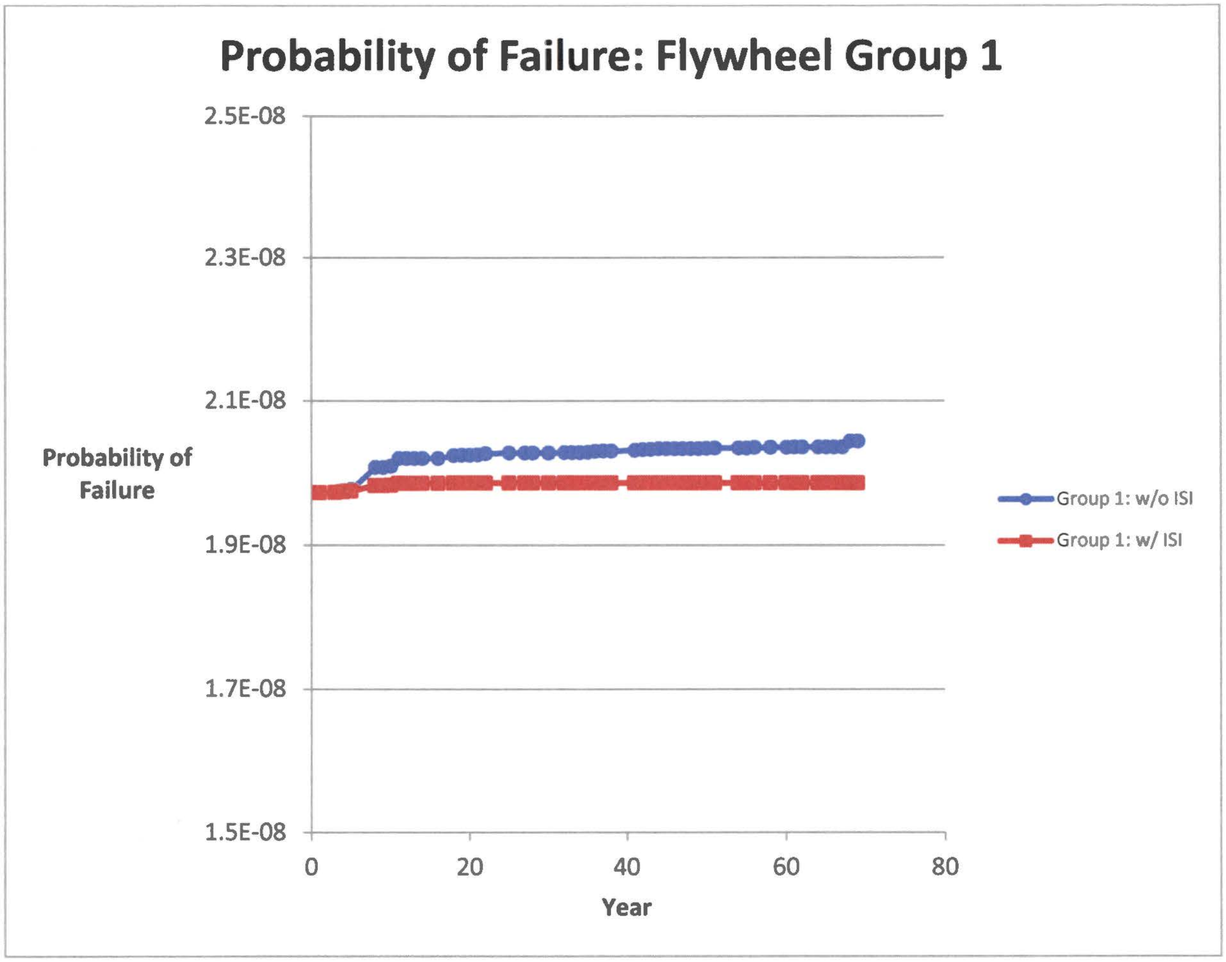

Figure 3-4: Probability of Failure for Flywheel Evaluation Group 1

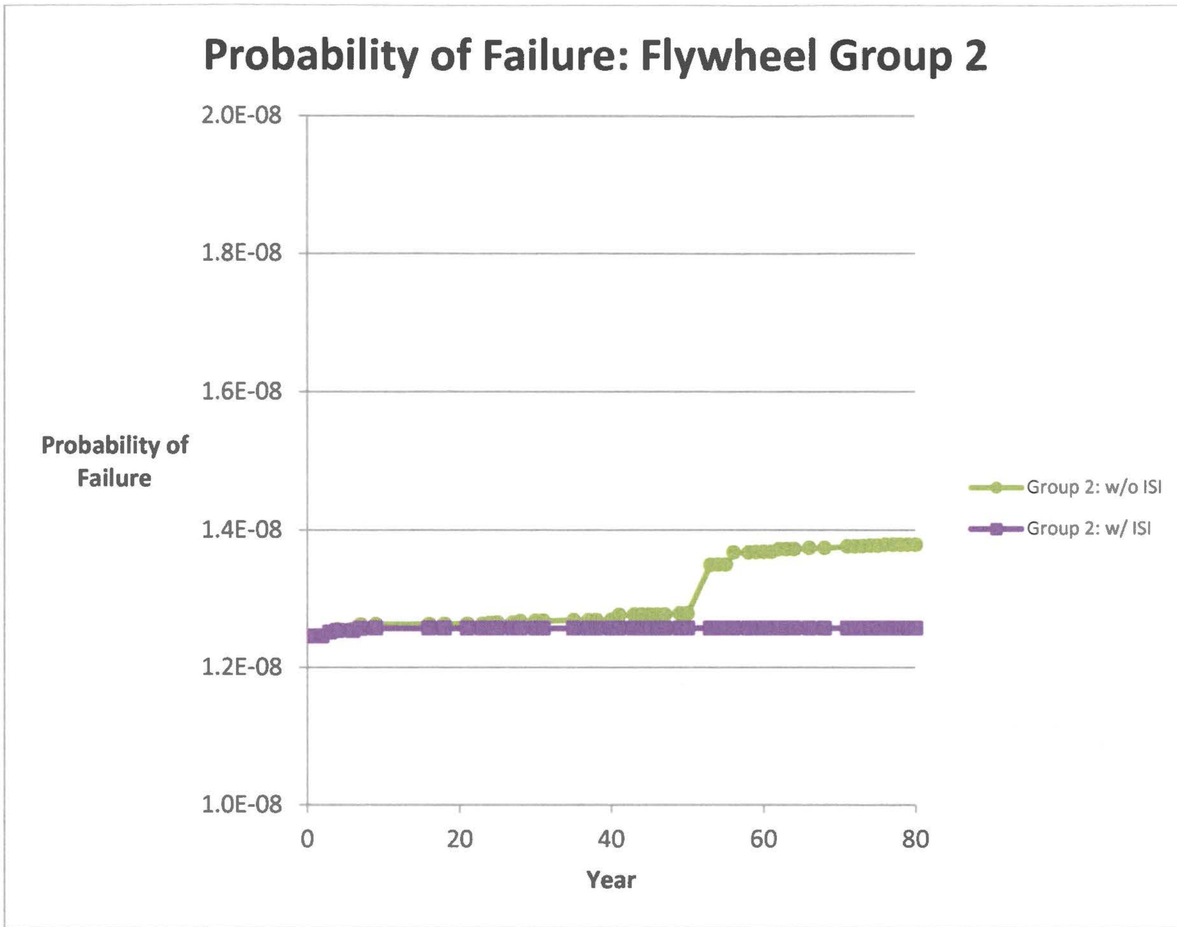

Figure 3-5: Probability of Failure for Flywheel Evaluation Group 2

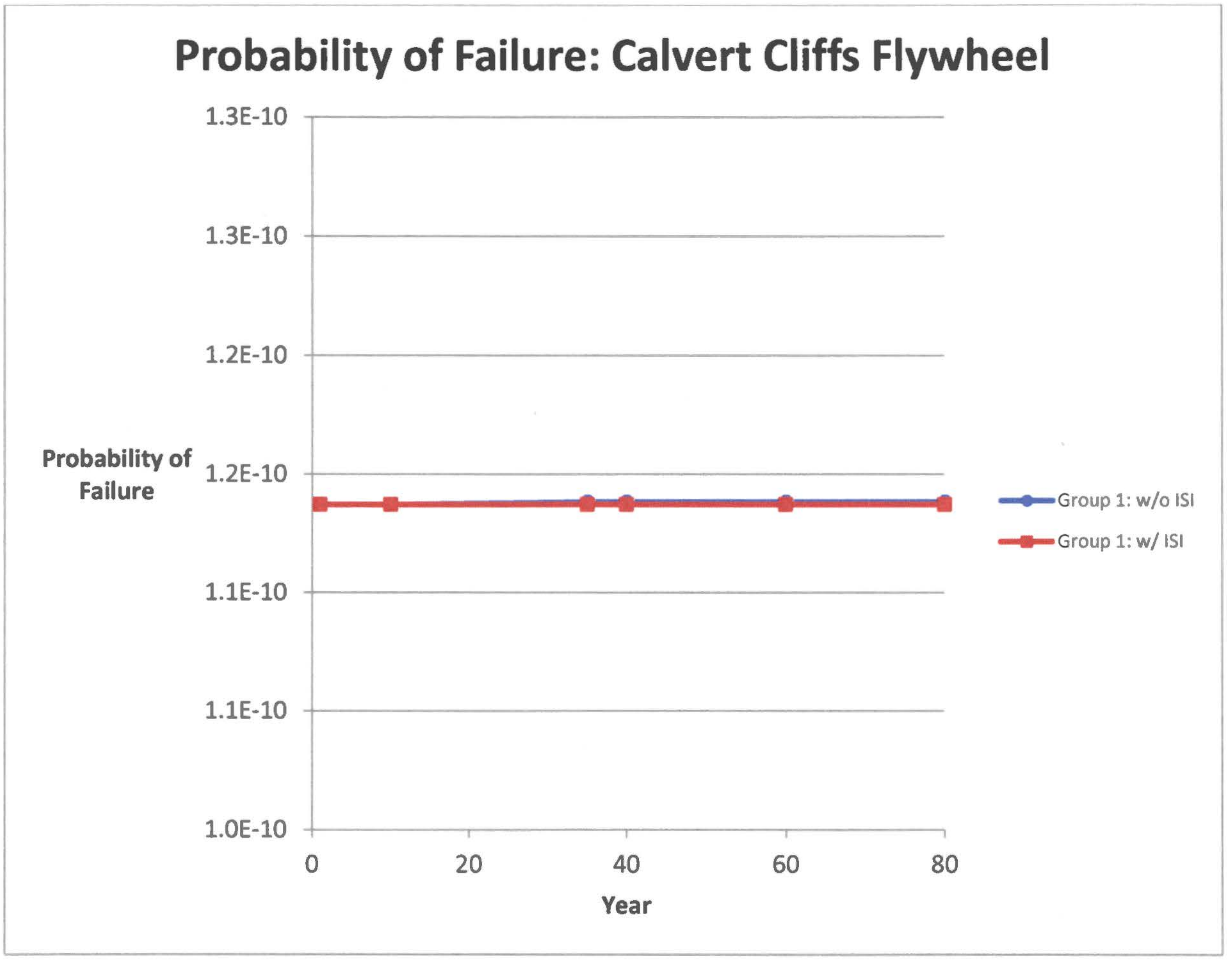

Figure 3-6: Probability of Failure for Calvert Cliffs Units 1 and 2

3.4 CORE DAMAGE EVALUATION

The objective of the risk assessment is to evaluate the core damage risk from the extension of the examination of the RCP motor flywheel, over an extended 80 year in-service duration, relative to other plant risk contributors through a qualitative and quantitative evaluation.

RG 1.174, Revision 2 [5] provides the basis for this evaluation and also provides the acceptance guidelines to make a change to the current licensing basis.

Risk is defined as the combination of likelihood of an event and severity of consequences of an event. Therefore, the following two questions are addressed:

- What is the likelihood of the event?
- What are the consequences?

The following sections discuss the likelihood and postulated consequences. The likelihood and consequences are then combined in the risk calculation and the results of the evaluation are presented.

Several different scenarios have been identified for potential RCP motor flywheel failures that are related to its operating speed and potential overspeed under certain conditions. These scenarios are summarized in Table 3-5.

Table 3-5: Summary of Flywheel Analysis Parameters

	Westinghouse RCP/Flywheel (rpm)	Calvert Cliffs RCP/Flywheel (rpm)
Failure during normal plant operation resulting in a plant trip	1500*	1125*
Failure of the RCP motor flywheel associated with a plant transient or LOCA event with NO loss of electrical power to the RCP	1500*	1125*
Failure of the RCP motor flywheel associated with a plant transient or LOCA event (up to a three square foot break in the main loop) with loss of electrical power to the RCP	1500*	1125*
Failure of the RCP motor flywheel associated with a large LOCA (from a greater than 3 ft2 break up to the DEGB of the RC loop piping) coincident with an instantaneous electrical power loss (e.g., loss of offsite power (LOOP) or loss of electrical power to the RCP) and therefore no electrical braking to the RCP	3321	1368

* Overspeed flywheel RPM for normal/accident conditions

3.4.1 What is the Likelihood of the Event

The likelihood is addressed by identifying a plant transient or LOCA event combined with the postulated failure of the flywheel and estimating the probability/frequency of these events. The likelihood of the flywheel failure is discussed in Section 3.3 and the results are provided in Table 3-3 for the two flywheel evaluation groups that bound the other flywheel groups and for the Calvert Cliffs Units 1 and 2 flywheels. The estimated failure probabilities for the different conditions for the various flywheel types and event combinations are shown in Table 3-6.

Table 3-6: Estimated RCP Motor Flywheel Failure Probabilities

Flywheel Group and Conditions*	Cumulative Probabilities of Flywheel Failure over 60 Years*		Cumulative Probabilities of Flywheel Failure over 80 Years*	
	With ISI at 4-Year Intervals	With ISI at 4-Year Intervals Prior to 10 Years and without ISI after 10 Years	With ISI at 4-Year Intervals	With ISI at 4-Year Intervals Prior to 10 Years, and without ISI after 10 Years
Group 1 – Normal/Accident*	1.99E-08	2.01E-08	1.99E-08	2.02E-08
Group 1 – LOCA/LOOP*	5.88E-02	5.88E-02	5.88E-02	5.88E-02***
Group 2 – Normal/Accident*	1.26E-08	1.36E-08	1.26E-08	1.37E-08
Group 2 – LOCA/LOOP*	1.66E-02	1.66E-02	1.66E-02	1.66E-02
Calvert Cliffs Units 1&2 – Normal/Accident**	1.14E-10	1.14E-10	1 14E-10	1.14E-10
Calvert Cliffs Units 1&2- LOCA/LOOP**	1.61E-07	1.63E-07	1.61E-07	1.63E-07

* For the failure probability calculations the mean flywheel speed for normal/accident conditions is 1500 rpm; for LOCA/LOOP is it 3321 rpm.

** For the failure probability calculations the mean flywheel speed for normal/accident conditions is 1125 rpm; for LOCA/LOOP is it 1368 rpm.

*** Selected as bounding value

3.4.2 What are the Consequences?

The consequence evaluation is performed to identify the potential consequences from the failure of the RCP motor flywheel from an integrity standpoint. The consequences are briefly discussed in Section 3.2.

The consequence evaluation includes both direct effects and indirect effects of a flywheel failure. Direct effects are those effects associated directly with the component being evaluated, such as loss of process fluid flow. Indirect effects are those effects on surrounding equipment that may be impacted by mechanisms such as jet impingement, pipe whip, missiles, and flooding.

The direct consequences are defined as failure of the RCP motor flywheel resulting in a failure of the RCP. If a failure of the RCP occurs, a reactor trip would result.

The potential indirect or spatial effects associated with the postulated flywheel failure are associated with the potential missiles generated from the fragmented portions of the flywheel associated with a significant flywheel crack.

For this evaluation, the conditional core damage probability associated with the failure of the flywheel will be assumed to be 1.0 (no credit for safety system actuation to mitigate the consequences of the failure).

3.4.3 Risk Calculation

This methodology is described in detail in WCAP-14572, Revision 1-NP-A, Supplement 1 [8]. For failures that cause only an initiating event, the portion of the PRA model that is impacted is the initiating event and its frequency. The core damage frequency from the failure is calculated by:

$$CDF = IE * CCDP_{IE}$$

Where:

CDF = Core Damage Frequency from a failure (events per year)

$CCDP_{IE}$ = Conditional Core Damage Probability for the Initiator

IE = Initiating Event Frequency (in events per year)

The initiating event frequency (in events per year) is obtained differently for the different conditions. For the normal operating mode, the initiating event frequency is determined from the RCP motor flywheel failure probability model as described in Section 3.3. Because the model generates a probability, the probability must be transformed into a failure rate. The cumulative probability at a given time is divided by the number of years to end of operating license. In other words,

$$IE = FP/EOL$$

where:

FP = Failure probability from failure probability model (dimensionless)

EOL = Number of years used in the failure probability model (80 years used to cover an extended plant life). Between 40 and 80 years, the failure probability is relatively constant.

For the RCP motor flywheel failure following an overspeed event, the core damage frequency of associated with that event (initiating event with flywheel failure) is defined as:

$$CDF = (IE * CFP) * CCDP$$

where:

CDF = Core Damage Frequency from a failure (events per year)

CCDP = Conditional Core Damage Probability for the initiator and flywheel failure

IE = Initiating Event frequency (in events per year)

CFP = Conditional Failure Probability of the flywheel by initiating event

The frequencies of the initiating events for the different conditions were identified as follows:

The initiating event frequency for a plant trip or non-LOCA transient is estimated as 1 event/year (plants on average experience 1 plant trip per year).

The probability of a loss of offsite power or loss of power to the RCP following a plant trip was conservatively established from NUREG/CR-6890 [13] as 0.01. This value was based on the observation that the conditional LOOP probability had increased from 0.003 in the 1986-1996 time frame to 0.0053 based on 1997 to 2004 data. Furthermore, the authors noted that the conditional probability in the summer months increase to 0.0091. The LOOP conditional on a LOCA event was estimated from Table 4.2 of NUREG/CR-6538 [12] as 1.4E-02 for PWR plants. LOCAs < 3 ft^2 and other plant transient events were conservatively combined, and the probability of a plant transient, concurrent with a LOOP, was conservatively represented by 0.014 and was used.

The frequency of a large break LOCA events with break areas in excess of 3 ft^2 (~23 inches in diameter) was estimated from NUREG-1829 [11]. Mean failure rates of piping are presented in Table 7.19 of that reference. Using 25 and 40 year failure rates, failure rates provided in that table were linearly extrapolated to 60 years and 80 years and then interpolated to obtain a mean frequency of exceeding 3.0 ft^2 , Using this process the LOCA exceedance frequency for break areas > 3 ft^2 was estimated to be approximately 3.8E-07 per year. For this analysis, the LOCA IE was assigned a bounding value of 1E-06 per year.

Table 3-7, Table 3-8, and Table 3-9 show the calculations that were used to estimate the frequency of the initiating event combined with the probability of the RCP motor flywheel failure. These calculations are also estimates of the core damage frequency given that the assumption of the CCDP is set to 1.0 (no credit taken for any safety systems).

The resulting calculations show that the change in CDF for flywheel Evaluation Group 1 is 3.57E-10/year/RCP, the change in the CDF for flywheel Evaluation Group 2 is 1.19E-09/year/RCP and the change in the CDF for the Calvert Cliffs flywheel is

1.05E-13/year/RCP as shown in Table 3-7, Table 3-8, and Table 3-9. The RG-1.174 criteria for an acceptable change in risk for CDF are 1E-06/year and for LERF is 1E07/year. These calculations show the change in risk from extending the inspection interval for the RCP motor flywheel is significantly below the acceptance criteria.

Table 3-7: Westinghouse RCP Motor Flywheel Evaluation Group 1

Condition	Initiating Event Frequency	Likelihood of RCP Motor Flywheel Failure (@80 years)		Event with RCP Motor Flywheel Failure (and Core Damage Frequency, CCDP = 1.0) (per year)	
	(per year)	With ISI after 10 Years	Without ISI after 10 Years	With IS After 10 Years	Without ISI after 10 Years
1 Normal Operating Condition	N/A	1.99E-08	2.02E-08	2.48E-10	2.53E-10
2. Failure of the RCP motor flywheel associated with a plant with NO loss of electrical power to the RCP at 1500 RPM.	1	1.99E-08	2.02E-08	1 99E-08	2.02E-08
3. Failure of the RCP motor flywheel associated with a plant transient (including LOCA event (up to a 3 ft^2 break in the RCS loop piping)) with loss of electrical power to the RCP (1200 rpm peak speed)** 1.0 x (1.4E-02)	1.40E-02	1.99E-08	2.02E-08	2.78E-10	2.83E-10
4. Failure of the RCP motor flywheel associated with a large LOCA (from a greater than 3 ft' break up to a DEGB of the RCS loop piping) coincident with an instantaneous power loss (e.g., loss of offsite power (LOOP) or loss of electrical power to the RCP) and therefore no electrical braking effects (3321 rpm peak speed)	1 40E-08	5.88E-02	5.88E-02	8.23E-10	8 23E-10
Totals				2.12E-08	2.16E-08
Change in CDF for one Flywheel (per RCP risk)					3 57E-10
Change in CDF for 4 RCPs (4 Flywheels)					1 43E-09

** 1500 rpm is used for the failure probability calculations

Table 3-8: Westinghouse RCP Motor Flywheel Evaluation Group 2

Condition	Initiating Event Frequency	Likelihood of RCP Motor Flywheel Failure (@80 years)		Event with RCP Motor Flywheel Failure (and Core Damage Frequency, CCDP = 1.0) (per year)	
	(per year)	With ISI after 10 Years	Without ISI after 10 Years	With IS After 10 Years	Without ISI after 10 Years
1. Normal Operating Condition	N/A	1.26E-08	1.37E-08	1.57E-10	1.72E-10
2. Failure of the RCP motor flywheel associated with a plant with NO loss of electrical power to the RCP at 1500 RPM	1	1.26E-08	1.37E-08	1 26E-08	1 37E-08
3. Failure of the RCP motor flywheel associated with a plant transient (including LOCA event (up to a 3 ft^2 break in the RCS loop piping)) with loss of electrical power to the RCP (1200 rpm peak speed)** 1.0 x (1.4E-02)	7.52E-09	1 26E-08	1.37E-08	1.76E-10	1 92E-10
4. Failure of the RCP motor flywheel associated with a large LOCA (from a greater than 3 ft^2 break up to a DEGB of the RCS loop piping) coincident with an instantaneous power loss (e.g , loss of offsite power (LOOP) or loss of electrical power to the RCP) and therefore no electrical braking effects (3321 rpm peak speed)	1.66E-02	1.66E-02	1.66E-02	2.32E-10	2.32E-10
Totals				1.31E-08	1.43E-08
Change in CDF for one Flywheel (per RCP risk)					1.19E-09
Change in CDF for 4 RCPs (4 Flywheels)					4 75E-09

** 1500 rpm is used for the failure probability calculations.

Table 3-9: Calvert Cliffs Units 1 and 2 RCP Motor Flywheel Evaluation

Condition	Initiating Event Frequency	Likelihood of RCP Motor Flywheel Failure (@80 years)		Event with RCP Motor Flywheel Failure (and Core Damage Frequency, CCDP = 1.0) (per year)	
	(per year)	With ISI after 10 Years	Without ISI after 10 Years	With IS After 10 Years	Without ISI after 10 Years
1. Normal Operating Condition	N/A	1 14E-10	1.14E-10	1 42E-12	1.42E-12
2 Failure of the RCP motor flywheel associated with a plant with NO loss of electrical power to the RCP 1200 RPM.	1	1.14E-10	1.14E-10	1 14E-10	1.14E-10
3 Failure of the RCP motor flywheel associated with a plant transient (including LOCA event (up to a 3 ft^2 break in the RCS loop piping)) with loss of electrical power to the RCP (900 rpm peak speed)** 1.0 x (1.4E-02)	1 40E-02	1.14E-10	1.14E-10	1.59E-12	1.59E-12
4. Failure of the RCP motor flywheel associated with a large LOCA (from a greater than 3 ft' break up to a DEGB of the RCS loop piping) coincident with an instantaneous power loss (e g., loss of offsite power (LOOP) or loss of electrical power to the RCP) and therefore no electrical braking effects (1368 rpm peak speed)	1.40E-08	1.61E-07	1.63E-07	2.25E-15	2.28E-15
Totals				1.17E-10	1.17E-10
Change in CDF for one Flywheel (per RCP risk)					1.05E-13
Change in CDF for 4 RCPs (4 Flywheels)					4.19E-13

3.5 CONSIDERATION OF UNCERTAINTY

This section provides a discussion of uncertainties associated with the core damage risk assessment. The discussion follows the general guidance of NUREG-1855 [14] in that the potential key model assumptions and uncertainties are identified and their impact is evaluated with respect to the current application.

The baseline risk assessment discussed in Section 3.4 includes several significant conservatisms which are intended to bias the results of the analysis in a conservative direction. Specifically, these assumptions include:

1. All flywheel failure events result in both core damage and a large early release This tacitly assumes that the missiles generated by the flywheel will result in both an unrecoverable LOCA and a loss in containment integrity sufficient to support a large release of radionuclides. This is a highly unlikely sequence as events resulting from a reactor trip would have control rods inserted prior to the failure and the potential for flywheel fragments to render all safety injection flow paths unavailable is unlikely. Furthermore, there is virtually no likelihood that flywheel fragments could significantly impact the ability of the containment to perform its function or prevent containment isolation.
2. The flywheel failure probability is based on a bounding selection of rotational flywheel speeds. This assumption is intended to simplify the event grouping while upwardly biasing the flywheel failure probabilities. The flywheel failure probability model used to assess the failure probability has been developed as a realistic model. Details of that model are provided in [2] and a sensitivity study to typical input assumptions is provided in Section 3.2.
3. Non-LOCA plant events that could result in a LOOP were assigned a LOOP probability of 0.014. This value is representative of the conditional LOCA/LOOP failure probability and as previously discussed overstates the LOOP potential for the more likely events.
4. The failure probabilities of flywheels are based on the cumulative failure probability over the lifetime of the flywheel. This is conservative because the failure rates are observed to stabilize during the later years of operation.
5. Probability of failure calculations assume crack growth is based on 100 flywheel start and stop cycles per year. This results in 8000 cycles for 80 years of operation. It is estimated that 6000 cycles will bound the lifetime operation of the flywheel, even when it is extended to 80 years of operation.

While these assumptions are intended to provide a bounding estimate of risk, uncertainty associated with other parameters may be of interest in understanding the potential risks of the risk evaluation. As discussed in Section 3.4, the risk of the inspection interval extension to 80 years has three elements: the frequency of the initiating event, the probability of flywheel failure associated with an event, and the conditional probability of core damage associated with the failure. Sensitivity studies were performed to investigate the potential impact in changes to the risk assessment

modeling assumptions. The results of these studies are included in Table 3-10. The uncertainty associated with each of these factors is discussed below.

3.5.1 Initiating Event Frequency

As discussed in Table 3-7 through Table 3-9 the flywheel failure risks are assigned to four bins: normal operation, plant transient events without a loss of off-site power, plant transient events (non-large LOCA) with a loss of off-site power and large LOCA events with a loss of offsite power. Normal operational events (for example RCP start-ups and shutdowns) are based on the flywheel operating life and a bounding number of start-up and shutdown cycles. This is a low contributor to flywheel failure risks. Transient events are assumed to result in acceleration of the flywheel to design speeds. The risk assessment assumed that the plant will experience one transient event per year. A review of plant operation in the United States between 1988 and 2015 demonstrates that overall plant operation has improved and more typical plant failure probabilities are less than 0.80 per year.[1] . The impact of the reduction in the plant trip frequency results in a 3.57E-10 per year per Group 1 RCP reduction in CDF from the baseline value shown in Table 3-10. The conditional LOOP probability contributes to the event frequencies for transient and LOCA events. Increasing the conditional LOOP probability from 0.014 to 0.05 only increases the incremental CDF by 3.70E-10 per year per RCP for Group 1 flywheels as shown in Table 3-10. Finally, the frequency of a large LOCA has a significant uncertainty attached to its mean value. In this study the large LOCA frequency (for breaks greater than 3 ft^2) was increased an order of magnitude from 1E-06 per year to 1E-05 per year with no observable impact on plant risk.

3.5.2 Conditional Flywheel Failure Probability

To simplify analyses flywheel failure probabilities were based on 80 year end of life failure assumption and, with the exception of the large LOCA event, the assumption that the flywheel failure condition occurs at the plant design flywheel speed. For Westinghouse plants this was 1500 rpm. However, many plant transients are expected to result in events with lower flywheel speeds closer to that of nominal operation. Assuming flywheel failure probabilities associated with 1200 rpm operation, per RCP core damage frequency would reduce to 5.28E-13 per year per Table 3-10.

3.5.3 Conditional Core Damage/Large Early Release Probability Associated with a Flywheel Failure Event

The baseline analysis assumes a conditional core damage probability and a conditional large early release probability of 1.0. As discussed above, this is a limiting assumption and the actual values are expected to be much lower; therefore, the conditional LERP probabilities would be negligible.

[1] IN/EXT-16-39534, Initiating Event Rates at U S. Nuclear Power Plants: 1988-2015, INL, May 2016.

Table 3-10: CDF Sensitivity to Variations in PRA evaluation assumptions for RCP Flywheel Failure Risk Assessment for Extending 10-year Inspection intervals to 80 years –(Flywheel Group 1)

	Incremental Change in CDF (per Year)	
	Risk Impact of Single Flywheel Failure	**Risk impact of Flywheel Failure (4 RCP Plant)**
Baseline Change in CDF	3 57E-10	1 43E-09
PWR general and other transient reduction to 0.8 per year	2.88E-10	1.15E-09
Increase the Conditional LOOP probability to 0 05	3.70E-10	1 48E-09
Increase the LOCA frequency for breaks >3 ft^2 to 1 E-05/year	3.57E-10	1.43E-09
Flywheel Failure probability reduced for normal operation and the non-large LOCA transient based on 1200 rpm	5.28E-13	2.11E-12

3.5.4 Conclusion Regarding Treatment of Uncertainty

The above sensitivity studies confirm that even for a relatively large increase in modeling parameters, the incremental CDF would continue to remain below the 1.0E-06 per year core damage and 1.0E-07 per year LERF criteria in [5] supporting the conclusion that this is a very small risk increase. This report assumes the incremental LERF and incremental CDF are equal. This is an extremely conservative assumption.

3.6 RISK RESULTS AND CONCLUSIONS

Given the extremely low failure probabilities for the RCP motor flywheel during normal/accident conditions and the extremely low probability of LOCA/LOOP, and assuming a CCDP of 1.0 (complete failure of the safety systems), the CDF and change in risk would still not exceed the risk criteria in [5] (ΔCDF<1.0E-6 per year and ΔLERF <1.0E-07 per year).

Even considering the uncertainties associated with this evaluation, the risk associated with the postulated failure of an RCP motor flywheel is significantly low. Even when all four RCP motor flywheels are considered in the bounding plant configuration case, the risk is still acceptably low.

Because of the evaluation results for core damage frequency and the conservative assumption that failure of the RCP motor flywheel results in core damage and a large early release, the calculations were not performed for the LERF. If detailed LERF analyses were performed, it is expected that the relative LERF contribution associated with these events would be significantly less than 20%. Regardless, this assessment assumes the calculated CDF is equal to LERF and that results are less than the LERF acceptance criterion (1E-07/reactor year).

The key principles identified in RG-1.174 were also reviewed and the responses based on the evaluation are provided in Table 3-11.

This evaluation, in conjunction with the previous deterministic calculations described throughout the report, concludes that the extension of the RCP motor flywheel examination from 10 to 20 years for RCP flywheels in operation up to 80 years would not be expected to result in a significant increase in risk; therefore, the proposed change is acceptable.

Table 3-11: Evaluation with Respect to Regulatory Guide 1.174 (Key Principles)

Key Principles	Evaluation Response
Change meets current regulations unless it is explicitly related to a requested exemption or rule change	No exemption or rule change is requested. This TR documents applicability of current ISI inspection intervals through 80 years of operation.
Change is consistent with defense-in-depth philosophy	The potential for failure of the RCP motor flywheel is negligible during normal accident conditions, and does not impact any plant structures, systems or components (SSCs).
Maintain sufficient safety margins	No safety analysis margins are changed.
Proposed increases in CDF or risk are small and are consistent with the Commission's Safety Goal Policy Statement	The proposed increase in risk is estimated to be negligible. The RCS leakage exists prior to a LOCA (no core damage consequences are associated with the RCS leakage) No credit taken is taken for RCS leakage detection
Use performance-measurement to monitor the change	NDE examinations are performed on a 20-year frequency for up to 80 years. Other indications of potential degradation of the RCP motor flywheel are available (e.g., pump vibration monitoring, and pump maintenance).

4 CONCLUSIONS

The results and conclusions as summarized in WCAP-14535A [1] remain valid and are reiterated below:

1. RCP flywheels are carefully designed and manufactured from excellent quality steel, which has a high fracture toughness.
2. The RCP flywheel overspeed is the critical loading; however, LBB has limited the maximum speed to 1500 rpm. *(Note, however, that LBB for LBLOCA was not considered in the risk assessment performed in WCAP-15666-A [2], which does consider the overspeed due to the LBLOCA.)*
3. RCP flywheel inspections have been performed for over 20 years, with no service-induced flaws.
4. The RCP flywheel integrity evaluations determined a very high flaw tolerance for the RCP flywheels
5. Crack growth during service is negligible.
6. The structural reliability studies concluded that eliminating inspections will not change the probability of failure.
7. The inspections result in man-rem exposure and the potential for flywheel damage during assembly and reassembly.

The deterministic results as summarized in WCAP-15666-A [2] remain applicable for 80 years of operation. The risk assessments are updated and presented in Section 3 of this report.

1. The failure probabilities for the RCP motor flywheels are small.
2. The change in risk is less than the Regulatory Guide 1.174 CDF and LERF acceptance criteria.
3. The 20-year ISI frequency for the RCP motor flywheel, approved by the NRC in [2], remains applicable for 80 years of operation.

5 REFERENCES

1. Westinghouse Report, WCAP-14535A, Rev. 0, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," November 1996.
2. Westinghouse Report, WCAP-15666-A, Rev. 1, "Extension of Reactor Coolant Pump Motor Flywheel Examination," October 2003.
3. United States Nuclear Regulatory Commission, Office of Standards Development, Regulatory Guide 1.14, Rev. 1, "Reactor Coolant Pump Flywheel Integrity," August 1975.
4. ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition with 2008 Addenda.
5. United States Nuclear Regulatory Commission, Regulatory Guide 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011.
6. United States Nuclear Regulatory Commission NUREG-0800, Standard Review Plan 19.0, "Use of Probabilistic Risk Assessment in Plant Specific Risk- Informed Decision Making: General Guidance."
7. F. J. Witt, "Development and Applications of Probabilistic Fracture Mechanics for Critical Nuclear Reactor Components," pages 55-70, Advances in Probabilistic Fracture Mechanics, ASME PVP-Vol. 92, 1984.
8. Westinghouse Report, WCAP-14572, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," Revision 1-NP-A, February 1999.
9. United States Nuclear Regulatory Commission NUREG/CR-5864, Theoretical and User's Manual for pc-PRAISE, A Probabilistic Fracture Mechanics Computer Code for Piping Reliability Analysis, July 1992.
10. *Documentation of Probabilistic Fracture Mechanics Codes Used for Reactor Pressure Vessels Subjected to Pressurized Thermal Shock Loading: Parts 1 and 2.* Electric Power Research Institute, Palo Alto, CA: June 1995. TR-105001.
11. United States Nuclear Regulatory Commission NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process," April 2008.
12. United States Nuclear Regulatory Commission NUREG/CR-6538, "Evaluation of LOCA With Delayed Loop and Loop With Delayed LOCA Accident Scenarios," July 1997.
13. United States Nuclear Regulatory Commission NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants: Analysis of Loss of Offsite Power Events," 1986-2004," December 2005.
14. United States Nuclear Regulatory Commission NUREG-1855, Rev. "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision making Final Report," July, 2016.
15. NRC Policy Issue SECY-82-465, "Pressurized Thermal Shock," 11/23/1982.

APPENDIX A: CALVERT CLIFFS UNIT 1 & 2 RCP MOTOR FLYWHEEL EVALUATIONS FOR EXTENSION OF ISI INTERVAL

Background and Purpose

WCAP-15666-A [2] extended the ISI intervals for Westinghouse RCP motors from 10 to 20 years. Although Calvert Cliffs Units 1 and 2 are Combustion Engineering NSSS plants, they have Westinghouse RCP motors and flywheels; however, the motor operating speeds are different than those evaluated in WCAP-15666-A [2]. A Calvert Cliffs plant-specific deterministic calculation and a probabilistic evaluation were performed using the methodology of [2] to justify 20-year ISI interval for 60 years of plant operation.

The probabilistic evaluations for Calvert Cliffs were updated in Section 3 of this report for 80 years of operation. The purpose of this Appendix is to evaluate and extend the applicability of [2] to 80-year plant operation for Calvert Cliffs Units 1 and 2.

Ductile Failure Analysis

As discussed in Section 2.3.2 of this report, the flywheel stresses are dependent on dimensions and rotation speed. Extending the operating period to 80 years does not affect the stress calculation. Therefore, the current ductile failure analysis for 60 years remains valid for 80 years of operation.

The ductile failure limiting speed was determined for the flywheel for two cases. Case 1 considered that no cracks were present but accounted for the reduced cross sectional area resulting from the keyway. Case 2 considered that a 10-inch radial crack existed emanating from the center of the keyway through the full thickness of the flywheel.

The calculated limiting speeds are:

Case 1: 3219 rpm (considering the keyway only, no crack)

Case 2: 2856 rpm (considering the keyway and a 10" crack)

Given the nominal operating speed of 900 rpm for Calvert Cliffs plants, criterion item f [3] is satisfied since this is lower than one half of the lowest calculated critical speed of 2856/2 = 1428 rpm, considering both no cracks present and a large crack (10") present.

Given the LOCA over speed of 1368 rpm for the Calvert Cliffs plants, criterion item f [3] is satisfied because it is less than any calculated critical speeds considering both no cracks present and a large crack present.

Non-ductile Failure Analysis

As discussed in Section 2.3.3 of this report, extending the operating period to 80 years does not affect the K_I calculations, and the flywheel fracture toughness, K_{Ic} would not change due to the 80 year extension. Therefore, the current non-ductile failure analysis for 60 years remains valid for 80 years of operation. As in discussed in Section 2.3.3, Table 2-5, RTNDT values of 0°F, 30°F and 60°F were used to calculate the critical flaw sizes shown in Table A-1.

Table A-1: Critical Crack Length in Inches and % Through Flywheel

RT_{NDT}	0°F	30°F	60°F
Critical Crack Length	18.5"	8.8"	3.7"
% Through the Flywheel	52%	25%	10%

Note: The % through the flywheel is calculated as CCL in the table divided by the radial length from the maximum radial keyway location to the flywheel outer radius [CCL / (41.0" - 4.7188" – 0.937")].

Fatigue Crack Growth

As discussed in Section 2.3.4 of this report, extending the operating period to 80 years does not affect the K_I and ΔK_I calculations. The 6000 design cycles of start and shutdown used for the FCG was determined to be bounding for 80 years of operation. However, the 6000 cycles for 80 years of operation must be confirmed for this TR to be applicable. The FCG of 0.025 inch after 80 years or 6000 cycles is negligible even when assuming a large initial crack length of 3.7 inches.

Excessive Deformation Analysis

As discussed in Section 2.3.5 of this report, the 80-year extension has no impact on the excessive deformation analysis of the flywheel. The current deformation results for 60 years remain applicable to 80 years of operation.

The change in the RCP flywheel bore radius and outer diameter at overspeed condition of 1368 rpm are:

Δa = the change in the flywheel bore radius at overspeed = 0.003 inch

Δb = the change in the flywheel outside radius at overspeed = 0.006 inch

Since Δ is proportional to ω^2, this represents a 231% increase $[(\omega_{os}/\omega_n)^2 = (1368 / 900)^2 = 2.31 = 231\%$ over the deformation at the normal operating speed.

This increase would not result in any adverse conditions, such as excessive flywheel vibrational stresses that would result in crack propagation since the flywheel assemblies are interference fit to the flywheel shaft and the calculated deformations are small and insignificant. It is noted that the deformation for Calvert Cliffs flywheels is less than the that of Westinghouse flywheels reported in Section 2.3.5 of this report.

Conclusion

The current Calvert Cliffs evaluation and results for 60 years are applicable for 80 years of operation. The stress and fracture evaluation results for Calvert Cliffs flywheels are consistent with the flywheels evaluated in [3]. The probabilistic risk evaluation, in conjunction with the deterministic calculations described above, concluded that extension of the RCP motor flywheel ISI from 10 to 20 years for flywheels in service up to 80 years is acceptable.

APPENDIX B: CORRESPONDENCE WITH U.S. NRC

The following pages are FPL responses to U.S. NRC RAIs regarding to the flywheel The first is the U.S. NRC SER reference 2, FPL letter to USNRC, on August 31, 2018 (ML18248A257). It has 18 attachments responding to different RAIs. Only Attachment 18 is relevant to the flywheel RAI No. 4.3.5-1, which is attached in this appendix. The second is the SER reference 3, FPL letter to USNRC December 21, 2018 (ML18362A146).

FPL

L-2018-152
10 CFR 54.17

August 31, 2018

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Re: Florida Power & Light Company
Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Turkey Point Units 3 and 4 Subsequent License Renewal Application
Safety Review Requests for Additional Information (RAI) Set 1 Responses

References:

1. FPL Letter L-2018-004 to NRC dated January 30, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application (ADAMS Accession No. ML18037A812)
2. FPL Letter L-2018-082 to NRC dated April 10, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application – Revision 1 (ADAMS Accession No. ML18113A134)
3. NRC RAI E-Mail to FPL dated August 6, 2018, Requests for Additional Information for the Safety Review of the Turkey Point Subsequent License Renewal Application – Set 1.(EPID No. L-2018-RNW-0002) (ADAMS Accession Nos. ML18218A199 and ML18218A200)

Florida Power & Light Company (FPL) submitted a subsequent license renewal application (SLRA) for Turkey Point Units 3 and 4 to the NRC on January 30, 2018 (Reference 1) and SLRA Revision 1 on April 10, 2018 (Reference 2).

The purpose of this letter is to provide, as attachments to this letter, responses to the safety review RAIs issued by the NRC on August 6, 2018 (Reference 3). Each RAI response and its corresponding attachment are indexed on page 2 of this letter. The attachments identify changes that will be made in a future revision of the SLRA (if applicable).

If you have any questions, or need additional information, please contact me at 561-691-2294.

A084
NRR

Florida Power & Light Company

700 Universe Boulevard, Juno Beach, FL 33408

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 31, 2018.

Sincerely,

William Maher
Senior Licensing Director
Florida Power & Light Company

WDM/RFO

Attachments: 18 RAI Responses (refer to Letter Attachment Index)

LETTER ATTACHMENT INDEX			
Attachment	NRC RAI	Attachment	NRC RAI
1	3.1.2.2.15-1	10	B.2.3.21-1
2	3.3.2.1.4-1	11	B.2.3.16-1
3	3.3.2.2.9-1	12	B.2.3.16-2
4	3.2.2.2.2-1	13	B.2.3.16-3
5	3.3.2.2.3-1	14	B.2.3.16-4
6	B.2.3.29-1	15	3.3.2.1.2-1
7	B.2.3.29-2	16	3.3.2.1.3-1
8	B.2.3.29-3	17	3.3.2.2.7-1
9	3.2.2.1.2-1	18	4.3.5-1

cc:

Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant
Regional Administrator, USNRC, Region II
Project Manager, USNRC, Turkey Point Nuclear
Senior Resident Inspector, USNRC, Turkey Point Nuclear
Plant Project Manager, USNRC, SLRA
Plant Project Manager, USNRC, SLRA Environmental
Ms. Cindy Becker, Florida Department of Health

NRC RAI Letter No. ML18218A199 Dated August 6, 2018

5. Reactor Coolant Pump Flywheel

Regulatory Basis:

10 CFR Section 54.21(c)(1) requires an applicant to provide a list of time-limited aging analyses (TLAAs) and demonstrate that (i) the analyses remain valid for the period of extended operation; (ii) the analyses have been projected to the end of the period of extended operation; or (iii) the effects of aging on the intended functions will be adequately managed for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR § 54.29(a)) is that actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under § 54.21, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis (CLB). The SRP-SLR provides the acceptance criteria for an applicant to demonstrate compliance with 10 CFR 54.21(c)(1). In order to complete its review and enable making a finding under 10 CFR § 54.29(a), the staff requires additional information in regard to the matters described below.

RAI 4.3.5-1

Background:

The Turkey Point SLR referenced PWROG-17011-NP, Revision 0 report, "Update for Subsequent License Renewal: WCAP-14535A, 'Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,' and WCAP-15666-A, 'Extension of Reactor Coolant Pump Motor Flywheel Examination'," dated May 2018 to support the TLAA for the Reactor Coolant Pump Flywheel. The PWROG-17011-NP, Revision 0 report provides the technical and regulatory basis for continuing the 20-year inservice inspection interval approved for reactor coolant pump motor flywheels for the 60 year period of license renewal in WCAP-15666-A into the 80 year period of subsequent license renewal.

Issue:

1. Section 3.4.3 of PWROG-17011-NP, Revision 0 provides descriptions and frequencies of the initiating events for the different conditions listed in Table 3-5. For the second condition, the description is, "The initiating event frequency for a plant trip or non-LOCA transient is estimated as 1 event/year (plants on average experience 1 plant trip per year)." This referencing of a non-LOCA transient does not appear in Tables 3-7 and 3-8 of PWROG-17011-NP, Revision 0 for condition 2. Further, it is contradictory to the description in Table 3-5 of PWROG-17011-NP, Revision 0 and in Tables 3-12 and 3-13 in WCAP-15666-A, which references a LOCA event: "Failure of the RCP motor flywheel given a plant transient or LOCA event with NO loss of electric power to the RCP."

2. Tables 3-7 to 3-9 of PWROG-17011-NP, Revision 0 show that the event frequency for the fourth condition is 1.4E-8/year. In WCAP-15666-A, the corresponding event frequency is 2.8E-8/year based on a maximum LOCA frequency (LOCAs with greater than 5000 gpm blowdown) of 2E-6/year and the probability of loss of station power following a LOCA of 1.4E-2/year. The staff previously approved the value for the fourth condition in WCAP-15666-A. The new event frequency is only half of this previously approved value. The staff needs to confirm the correct value to determine the change in CDF and LERF to verify within the acceptable range in Regulatory Guide 1.174.

Request:

1. Please provide clarification for the inconsistencies between Tables 3-5, 3-7, 3-8, 3-12, and 3-13 regarding LOCA events vs. non-LOCA events.
2. Please justify the difference in the event frequency for the fourth condition between PWROG-17011-NP and WCAP-15666-A.

FPL Response:

1. Table 3-5 of PWROG -17011-NP lists several different conditions that have been identified for potential Reactor Coolant Pump (RCP) motor flywheel failures that are related to its operating speed and potential overspeed under certain conditions. Condition 2 on this listing is "Failure of the RCP motor flywheel given a plant transient or LOCA event with NO loss of electrical power to the RCP." A key attribute of this Condition that pertains to Tables 3-7 and 3-8 is the "NO loss of electrical power to the RCP," as this condition maintains the rpm of the flywheel for the event.

   This condition is carried over to Tables 3-7 and 3-8 as the listed Condition 2, and also considers all transients (both LOCA and non-LOCA) with no loss of electrical power to the RCP. The transient frequency contribution of 1 per year as listed in Condition 2 of Tables 3-7 and 3-8 is intended to subsume the non-large LOCA events. As noted in PWROG-17011-NP, these non-large LOCAs are LOCAs with break sizes of less than 3ft^2. The flywheel performance for these events will be bounded by the design limiting transient. The LOCA contribution is negligible as its frequency contribution is on the order of [0.001] per year and no change in the calculation would result as the frequency remains bounded by 1 per year.

   This Condition 2 in PWROG-17011-NP is also the same as the 'Condition 2' entries in Tables 3-12 and 3-13 of WCAP-15666-A. All entries represent the same conditions and employ the same analysis assumptions.

2. The frequencies of large break LOCA events with break areas in excess of 3 ft^2 (fourth condition) reported in WCAP-15666 were based on Westinghouse fracture mechanics calculations performed prior to NRC issuance of NUREG-1829 (Reference 1). The frequencies of large break LOCA events with break areas in excess of 3 ft^2 estimated in PWROG-17011-NP were updated based on NUREG

1829 and the mean failure rates associated with the larger LOCA break sizes presented in Table 7.19 of this Reference.

Reference 1 states that "The results in Table 7.19 are appropriate to use for PRA applications that separately consider SGTRs." Specifically, in establishing the large break frequency, the 14 inch and 31 inch diameter breaks were extrapolated to 80 years and interpolated to determine a cumulative frequency for a 3 ft^2 break. NUREG-1829 was used as it is consistent with current probabilistic risk assessment practices and associated Regulatory Guide 1.174 submittals.

References:

1. NUREG-1829 Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process: Main Report, USNRC, March 2008

Associated SLRA Revisions:

No SLRA changes have been identified as a result of this response.

Associated Enclosures:

None

FPL

L-2018-234
10 CFR 54.17

December 21, 2018

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Re: Florida Power & Light Company
Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety
Review Request for Additional Information (RAI) Set 5 Response 4.3.5-2 Revision

References:

1. FPL Letter L-2018-004 to NRC dated January 30, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application (ADAMS Accession No. ML18037A812)

2. FPL Letter L-2018-082 to NRC dated April 10, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application – Revision 1 (ADAMS Accession No. ML18113A134)

3. FPL Letter L-2018-175 to NRC dated October 17, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application – Safety Review Requests for Additional information (RAI) Set 5 Responses (ADAMS Accession No. ML18292A642)

On April 10, 2018, Florida Power & Light Company (FPL) submitted to the NRC Revision 1 of the subsequent license renewal application (SLRA) for Turkey Point Units 3 and 4 (Reference 1), as well as supplemental information for the SLRA Environmental Report (ER) (Reference 2).

The purpose of this letter is to provide, as an attachment to this letter, the revised response to safety review Set 5 RAI 4.3.5-2 regarding the conditional probability of failure (PoF) for reactor coolant pump (RCP) flywheels. This revised response supersedes the corresponding RAI response provided in Reference 3.

If you have any questions, or need additional information, please contact me at 561-691-2294.

A084
NKR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 21, 2018.

Sincerely,

William Maher
Senior Licensing Director
Florida Power & Light Company

WDM/RFO

Attachment: FPL Revised Response to NRC RAI No. 4.3.5-2

cc:

Senior Resident Inspector, USNRC, Turkey Point Plant
Regional Administrator, USNRC, Region II
Project Manager, USNRC, Turkey Point Plant
Plant Project Manager, USNRC, SLRA
Plant Project Manager, USNRC, SLRA Environmental
Ms. Cindy Becker, Florida Department of Health

NRC RAI Letter Nos. ML18260A242 and ML18260A243 dated September 17, 2018

RAI 4.3.5-2

Background:

To support its TLAA disposition of § 54.21(c)(1)(i), the applicant included the PWR Owners Group report, PWROG-17011-NP, Rev. 0, August 2017 in Enclosure 4 of the SLRA. PWROG-17011-NP provides the generic SLR methodology for deterministic and risk-informed analyses related to integrity of Westinghouse RCP motor flywheels. PWROG-17011-NP is not approved by the NRC for use in SLR applications. In order to complete its review of this TLAA the staff must determine whether the applicant's proposed implementation of the generic SLR flywheel methodology in PWROG-17011-NP is acceptable for demonstrating, per § 54.21(c)(1)(i), that the CLB analyses of the PTN 3 and 4 flywheels will remain valid for the subsequent period of extended operation (SPEO).

Issue:

The risk assessment in PWROG-17011-NP, Section 3 used the probabilistic fracture mechanics (PFM) analysis methodology to generate conditional probability of failure (PoF) for reactor coolant pump (RCP) flywheels for the 80-year risk assessment. The staff noted that the PoFs for the case when ISI was performed every 4 years for only the first 10 years of operation in PWROG-17011-NP for 80-year SLR terms are lower than the corresponding PoFs in WCAP-15666-A for 60-year initial LR terms. The reason for this is not clear. In theory, when a selected flaw is given 20 more years to grow without any ISI, the PoF should be higher.

Request:

Please explain why the PoFs for the ISI case documented above are lower for the 80-year SLR analysis in PWROG-17011-NP compared to the corresponding PoFs for this ISI case for the 60 year LR analysis in WCAP-15666-A.

If there is an error in the PoF analysis for this ISI case, please provide the correct PoF calculations for the 60-year and/or 80-year analyses, as needed. Please revise the PWROG-17011-NP report, as needed, to show the correct PoF calculations.

FPL Revised Response:

This revised RAI response supersedes in its entirety the RAI response provided in Attachment 1 of Reference 6 [6].

PWROG-17011-NP [1] is a generic topical report that supports reactor coolant pump (RCP) Flywheel 20-year inspection intervals for plants with Westinghouse original equipment manufacturer (OEM) RCPs. This report is an update to WCAP-14535A (40-year) [2] and WCAP-15666-A (60-year) [3], both reports receiving a Safety Evaluation Report (SER) from the Nuclear Regulatory Commission (NRC), as PWROG-17011 extends the evaluation period of the flywheel analysis to 80 years of operation. This

work was done under the purview of the Owners group and performed to support the fleet as plants begin to apply for extending their license from 60 to 80 years.

As in the 40 and 60 year evaluations in [2 and 3], the analysis performed in PWROG-17011 consists of both a deterministic stress and fracture evaluation and a probabilistic risk assessment. The risk assessment is comprised of two parts: probabilistic fracture mechanics (PFM) evaluation to determine probability of RCP flywheel failure, and a core damage evaluation. The PFM analyses used a Westinghouse developed computer program, RPFWPROF, that calculates the difference in RCP flywheel failure probabilities with and without In-Service Inspections (ISI). All three reports, WCAP-14535A, WCAP-15666-A, and PWROG-17011-NP include both the deterministic and probabilistic analyses. The NRC's SER for WCAP-14535A was solely based on the deterministic evaluation and granted an ISI interval of 10 years. The NRC's SER for WCAP-15666-A reviewed both the deterministic and probabilistic evaluation and granted an ISI interval of 20 years.

In seeking to address the NRC's RAI above, Westinghouse engineers determined that the RPFWPROF executable file used in PWROG-17011-NP cannot reproduce the results of WCAP-15666-A when run on original computer platforms. This confirms that Westinghouse had lost configuration control on the PFM executable program, RPFWPROF.

To correct for this, Westinghouse has performed a review of the deterministic aspects of the analysis to ensure their continued appropriateness and then also re-established configuration control of the RPFWPROF program and determined revised RCP flywheel probabilities of failure for 40, 60, and 80 years of operation.

Engineering Assessment of the Deterministic Evaluation

The function of the RCP in the reactor coolant system (RCS) of a pressurized water reactor plant is to maintain an adequate cooling flow rate by circulating a large volume of primary coolant water at high temperature and pressure through the RCS. A concern over overspeed of the RCP and its potential for failure led the NRC to issue Regulatory Guide (RG) 1.14 [5] in 1975. RG 1.14, Revision 1, Section C, subsection 2, provides the NRC regulatory guidance for flywheel design. Key provisions of this position are to analyze for ductile failure, analyze for non-ductile failure, demonstrate compliance with excessive deformation failure criterion, and demonstrate compliance with loss of coolant accident (LOCA) overspeed criterion.

WCAP-14535A

Since the NRC granted the 10-year ISI interval based solely on the 40-year deterministic analyses in WCAP-14535A [2], the loss of configuration control of the RPFWPROF program does not affect the NRC staff decision. The deterministic evaluation contained in the report remains valid and meets requirements of RG 1.14.

WCAP-14535A classifies flywheels into 16 groups according to their material and geometric information, and selected six flywheel groups for evaluation, which encompass the range of domestic flywheel dimensions covered by the report. In Section 4.2 of the report [2], a linear elastic stress analysis was performed and demonstrates that the flywheel structure can resist ductile failure with sufficient margin of safety during faulted conditions by meeting the faulted condition criteria of ASME B&PV Code Section III. The calculated critical speed due to ductile failure satisfied the RG 1.14, Section C, items 2.f for normal speed of 1200 revolutions per minute (RPM), and item 2.g, LOCA overspeed of 1500 RPM. The analysis performed and documented in WCAP-14535A to evaluate for ductile failure has been reviewed and confirmed to remain valid.

To evaluate for non-ductile failure, the analysis used the closed-form solution for a radial full-depth crack emanating from the bore of a rotating disk to calculate the applied stress intensity factor. The fracture toughness, K_{IC} as a function of flywheel temperature and RT_{NDT}, was calculated per the ASME Code, Section XI, A-4200, a lower bound curve. The assumed operating temperature of 70°F is conservative because the typical containment ambient temperature is 100°F to 120°F. Furthermore, the flywheel would receive additional heat from the pump motor due to friction and from reactor coolant through conduction through the pump itself. RT_{NDT} was assumed to be 60°F for additional conservatism as the 1973 equipment specification requires RT_{NDT} no greater than 10°F. The loads used in calculating the applied K_I were from an overspeed of 1500 RPM. As shown in Table 4-3 of [2], the critical crack length for flywheel overspeed of 1500 RPM are large, even considering the conservative flywheel temperature and RT_{NDT}. As noted in the SER, the original submittal did not calculate a critical speed based on an assumed initial flaw as requested by RG 1.14. In response to this, allowable crack sizes were calculated in Table 1of [2] using the margins in IWB-3610 of Section XI of the ASME Code, and including the effect of shrink fit. The analysis performed and documented in WCAP-14535A to evaluate for non-ductile failure has been reviewed and confirmed to remain valid.

To demonstrate compliance with excessive deformation failure criterion, the analysis in [2] used standard closed-form formulas for rotating disks to calculate the change of flywheel inner and outer radii at 1500 RPM. Results presented in Table 4-5 of [2] demonstrate a maximum flywheel deformation of only 0.010 inches and conclude that this would not result in adverse conditions since the flywheel assemblies are typically shrunk fit to the flywheel shaft and the deformations are negligible. The analysis performed and documented in WCAP-14535A to evaluate for compliance with excessive deformation failure criterion has been reviewed and confirmed to remain valid.

WCAP-15666-A

The purpose of WCAP-15666-A [3] is to extend the ISI interval from 10 years [2] to 20 years for a 60 year operating lifetime. It was determined that Groups 1 and 2 bound the

11 groups evaluated in [2]. The deterministic fracture mechanics evaluation used the same methodology in [2] to demonstrate acceptance of RG 1.14 requirements regarding ductile failure, nonductile failure, fatigue crack growth and excessive deformation. The NRC staff agreed that RG 1.14 requirements were satisfied in the SER. Although the loss of configuration control of the PFM program RPFWPROF affects WCAP-15666-A, the PFM evaluation is completely separate from the deterministic stress and fracture mechanics. The RG 1.14 requirements were satisfied solely based on the deterministic evaluations. Therefore, there is no impact on the RCP flywheels from performing their intended safety function for 60 years of operation.

PWROG-17011-NP

The purpose of PWROG-17011-NP [1] is to extend the evaluation and conclusion in [3] for an extended operating period of 80 years. The only time dependent variable of the deterministic evaluation is the number of cycles of RCP start and shutdown. It was confirmed that the assumed 6000-cycle in [3] remains conservative and unchanged going from 60 to 80 years. Additionally, since the flywheel is not local or adjacent to the reactor core, RT_{NDT} remains unchanged for 80 years. Therefore, the deterministic evaluation results and conclusion in WCAP-15666-A [3] remains applicable for [1]. The RG 1.14 requirements were satisfied, and the RPFWPROF program issue has no impact on the RCP flywheels from performing their intended safety function for 80 years of operation.

Review of Fleet Operating Experience

In preparing [2], in 1995, a review was performed and identified 729 flywheel inspections have been performed, with only 43 resulting in recordable indications. None of these indications were determined to affect flywheel integrity.

A review of industry operating experience was also performed in 2017 and reported in [1] as part of the work in extending the evaluation to 80 years. Since the inspection records reported in [2], there are 81 flywheel inspection records with only 4 resulting in recordable indications. All 4 of these indications were determined to be non-relevant to flywheel integrity, and are attributed to the disassembly and reassembly during inspection.

Conclusions from Deterministic Evaluation Review

The deterministic work performed in [1 to 3] have been reviewed and confirmed. The deterministic work demonstrates that requirements of RG 1.14 have been satisfied with significant margins, ensuring the reactor can be safely shutdown, the core will be cooled per the LOCA licensing criteria, and the reactor will remain shut down. Additionally, significant fleet operating experience with flywheel inspections demonstrates, empirically, no indications to date that shown to have impacted RCP flywheel integrity.

Overview of Investigations into Probabilistic Fracture Mechanics Program, RPFWPROF

As discussed above, the RAI questions the results (probabilistic results are ~3% less than the WCAP-15666-A work) and requests FPL and Westinghouse to confirm that there are no errors in what was submitted. As a note: the 2003 report (WCAP-15666-A), listed Probability of Failures (POF) at 2.57E-7 for Flywheel Group 1 flywheels at 60 years (with ISI at 4 year intervals prior to 10 years and without ISI after 10 years) while PWROG-17011-NP shows POF at 2.49E-7 at 60 years; and 2.52E-7 at 80 years for this same flywheel grouping and same ISI assumptions.

Westinghouse had noted the small difference in predictions between the 40 and 60 year cases between the two reports. At the time these differences were attributed to a change in computer platforms and a full set of interval calculations were performed with the available version of the RPFWPROF so that delta interval calculations would be self-consistent. In seeking to address the NRC's current RAI confirming Westinghouse's initial judgement and regarding the K_{IC} model, Westinghouse engineers determined that the RPFWPROF executable file used in PWROG-17011-NP cannot reproduce the results of WCAP-15666-A when run on original computer platforms. This confirms that Westinghouse had lost configuration control on the PFM executable program, RPFWPROF.

The RPFWPROF program was not re-written, but was used in what was believed to be the same form as that used in the 60 year POF analysis in the determination of the POF for 80 years [3]. In support of the PWROG-17011-NP report, the engineers retrieved the RPFWPROF executable file from local engineering group's shared folder, ran the RPFWPROF executable program, and compared RCP probability of failure results with those in WCAP-15666-A. Because the results were within ~3% of WCAP-15666-A result, and the 80 year POF was greater than the 60 year POF, demonstrating that the POF would increase with time as expected, the engineers considered the ~3% difference in results to be reasonable for a probabilistic evaluation and attributed the POF difference to the employment of different (upgraded) computer platforms between the work performed in 2003 and 2017.

In seeking to address the NRC's current RAI, and in particular the question regarding the basis for selected K_{IC} values in the probabilistic assessment, an error was uncovered in the available hard copy of the original independently reviewed source code for RPFWPROF. Therefore, the assessment of reasonableness was reviewed more closely, and Westinghouse engineers attempted to reproduce the original results of the probabilistic fracture mechanics program of WCAP-15666-A, RPFWPROF when removing the computer platform differences.

As part of this reproduction, benchmarking cases that were used to test the original RPFWPROF program were re-performed now, using the RPFWPROF executable program used for the PWROG-17011-NP work, but running the executables on older computer platforms. The purpose of this reproduction was to confirm that the

RPFWPROF executable program used in support of the PWROG-17011-NP report matched that used in the earlier reports – (removing the computer platform difference as a variable). The benchmarking case results could not reproduce the results from [3]; therefore, this confirms that Westinghouse has lost configuration control on the probabilistic fracture mechanics executable program, RPFWPROF.

Re-establishing Configuration Control of the RPFWPROF Program

The RPFWPROF executable files were re-constructed to follow the original methodology. Additional lines of code were added to allow for validation of intermediate functional steps of the program, but all other aspects of the program were purposely left unchanged to ensure conformity with the RPFWPROF program detailed in Section 3.3 of [3]. Thus, parameter values listed in Tables 3-1 through Tables 3-6 of [3] remain unchanged. Results of the program were validated using hand calculations to ensure that the program was appropriately considering the physical phenomena being simulated, and also validated by making comparisons to the original program's benchmarking case results, as appropriate, as these are allowed validation methods in Westinghouse's validation and verification program for programs. The result of this effort is a revised RPFWPROF program that has been confirmed to appropriately calculate the probability of RCP Flywheel failures.

By this verification and validation process, using the 1995 independently reviewed version of RPFWPROF, the POF results listed in Table 3-8 of WCAP-15666-A [3] can be exactly reproduced. However, it has been determined that there is an error in the RPFWPROF program that supports the WCAP-15666-A report. The program compares a stress intensity factor, K_I, for a set of input conditions, to the fracture toughness, or critical stress intensity factor, K_{IC}, and where $K_I > K_{IC}$, the flywheel is considered to have failed for that set of conditions. K_{IC} is calculated in the RPFWPROF program using the mean curve developed through regression analysis and documented in [4], as shown below.

$$\bar{K}_{Ic} = 55.1 + 28.0\exp(0.0214(T-RT_{NDT})) \quad \text{for } T-RT_{NDT} > -50°F \quad \text{(H-5b)}$$

During verification and validation efforts, it was determined that there was a transcription error in the program code and the exponent 0.0214 was mistakenly written as 0.2014. This results in K_{IC} increasing rapidly to the upper limit of 200 ksi$\sqrt{in}$ as the difference of service temperature minus the reference temperature for nil ductility transition ($T-RT_{NDT}$) increases. For positive differences in ($T-RT_{NDT}$) this error results in a higher allowable when determining whether a failure occurred. The opposite is true for predicted conditions with negative differences in ($T-RT_{NDT}$). Thus, while the WCAP-15666-A results can be exactly reproduced, the calculated POFs must be corrected to account for this error.

It should also be noted that the RPFWPROF executable program ran in 2017 to support issuance of PWROG-17011-NP, Rev. 1 [1] is being discarded. While results from this executable program were within 3% of the WCAP-15666-A POF results, Westinghouse's present verification and validation efforts have determined that changes were made to the RPFWPROF executable program utilized for the PWROG-17011-NP, Rev. 1 report following the work performed in support of WCAP-15666-A and were not managed through configuration control, resulting in calculations that should be discarded.

Revised RCP Flywheel POFs

Cumulative Probability of Failure over 40, 60, and 80 Years With and Without Inservice Inspections *(Values from Table 3-8 of WCAP-15666-A are shown italicized in parentheses below each newly calculated value for ease of comparison)*					
Flywheel Group	Design Limiting Speed (RPM)	Cumulative Probability of Flywheel Failure with ISI at 4-year Intervals	Cumulative Probability of Flywheel Failure with ISI at 4-year intervals prior to 10 Years, and without ISI after 10 years		
		Over 80 Years	Over 40 Years	Over 60 Years	Over 80 Years
1	1500	1.99E-08 *(2.45E-07)*	2.00E-08 *(2.50E-07)*	2.01E-08 *(2.57E-07)*	2.02E-08
2	1500	1.26E-08 *(1.43E-07)*	1.26E-08 *(1.45E-07)*	1.36E-08 *(1.47E-07)*	1.37E-08
1	3321	5.88E-02 *(1.01E-02)*	5.88E-02 *(1.01E-02)*	5.88E-02 *(1.02E-02)*	5.88E-02
2	3321	1.66E-02 *(0.91E-02)*	1.66E-02 *(0.91E-02)*	1.66E-02 *(0.91E-02)*	1.66E-02

As shown in WCAP-15666-A, Table 3-7, the 60-year analysis used a sample size of 9,999. Due to the lower POF results of the 80-year analysis, the 80-year analysis used a sample size of 99,999 to ensure statistical stability of the POF results.

Conservatisms In the Analysis

The probabilistic fracture mechanics evaluation, RPFWPROF, conservatively assumes that the probability of a flaw existing after preservice inspection is 10% and that the ISI flaw detection probability is only 50%. In reality, most pre-service and ISI flaws would be detected, especially for the large flaw depths that may lead to failure.

Furthermore, the core damage evaluation performed for the analysis considers the double-ended guillotine break (DEGB) coincident with instantaneous loss of offsite power event (3321 RPM with uncertainties that allows variations about 3000 to 3600 RPM) leading to no electrical braking to the RCP, resulting in likelihood of core damage to be well within RG 1.174 [5] guidelines. The risk assessment does not take credit for the leak before break (LBB) strategy that would easily detect and eliminate breaks, which would limit the maximum LOCA overspeed to 1500 RPM per the deterministic evaluations in [2 and 3].

During an accident like the DEGB, the containment temperature would be much higher than the currently assumed flywheel temperatures. As discussed in [2 and 3], the flywheel equipment specifications require both Charpy and drop-weight tests to ensure the flywheel RTNDT to be no greater than 10°F. Therefore, the RPFWPROF input for RT_{NDT} of 30°F as the median with 17°F uncertainty is conservative.

Conclusion

The probabilistic fracture mechanics work performed in [1] has been reviewed and corrections are being made to [1]. The computer program, RPFWPROF, has been verified and validated through Westinghouse's software configuration control process and the corrected POFs are provided above. No other Westinghouse documents are affected by the transcription error in the RPFWPROF executable program.

References:

1. PWROG Report, PWROG-17011-NP, Rev. 1 "Update for Subsequent License Renewal: WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination" and WCAP-15666-A, "Extension of Reactor Coolant Pump Motor Flywheel Examination," May 2018.
2. Westinghouse Report, WCAP-14535A, Rev. 0, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," November 1996.
3. Westinghouse Report, WCAP-15666-A, Rev. 1, "Extension of Reactor Coolant Pump Motor Flywheel Examination," October 2003.
4. NRC Policy Issue SECY-82-465, "Pressurized Thermal Shock," 11/23/1982 (ADAMS Accession No. ML16232A574)
5. NRC Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity," August 1975 (ADAMS Accession No. ML003739936)
6. FPL Letter L-2018-175 to NRC dated October 17, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application, Safety Review Requests for Additional Information (RAI) Set 5 Responses (ADAMS Accession No. ML18292A642)

Associated SLRA Revisions:

None

Associated Enclosures:

None