NEXTera ENERGY
DUANE ARNOLD

April 20, 2017

NG-17-0084
10 CFR 50.90

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

Duane Arnold Energy Center
Docket No. 50-331
Renewed Facility Operating License No. DPR-49

License Amendment Request (TSCR-164), Revision to Technical Specification 3.1.2, Reactivity Anomalies

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), NextEra Energy Duane Arnold, LLC (NextEra) is submitting a request for an amendment to the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC). The proposed amendment would revise TS 3.1.2, "Reactivity Anomalies," with a change to the method of calculating core reactivity for the purpose of performing the reactivity anomaly surveillance.

The Enclosure to this letter provides NextEra's evaluation of the proposed change. Attachment 1 to the enclosure provides a markup of the TS showing the proposed changes, and Attachment 2 provides the proposed TS Bases changes. The changes to the TS Bases are provided for information only and will be incorporated in accordance with the TS Bases Control Program upon implementation of the approved amendment.

NextEra requests approval of the proposed license amendment by May 1, 2018, and implementation within 90 days.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated State of Iowa official.

As discussed in the Enclosure, the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the change. The Duane Arnold Energy Center Onsite Review Group has reviewed the proposed license amendment request.

This letter contains no new or revised regulatory commitments.

If you have any questions or require additional information, please contact Michael Davis, Licensing Manager, at 319-851-7032.

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

Executed on April 20, 2017.

P Hansen for Dean Curtland

Dean Curtland
Site Director, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Enclosure

cc: Regional Administrator, USNRC, Region III,
Project Manager, USNRC, Duane Arnold Energy Center
Resident Inspector, USNRC, Duane Arnold Energy Center
A. Leek (State of Iowa)

ENCLOSURE 1 to NG-17-0084

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

License Amendment Request (TSCR-164), Revision to Technical Specification 3.1.2, Reactivity Anomalies

EVALUATION OF PROPOSED CHANGE

ATTACHMENT 1 - Markup of Technical Specifications
ATTACHMENT 2 - Markup of Technical Specification Bases

6 pages follow

1.0 SUMMARY DESCRIPTION

NextEra Energy Duane Arnold, LLC (NextEra) hereby requests an amendment to the Duane Arnold Energy Center (DAEC) Technical Specifications (TS) to modify TS 3.1.2, "Reactivity Anomalies." The proposed change would allow performance of the reactivity anomaly surveillance on a comparison of monitored to predicted core reactivity. The reactivity anomaly verification is currently determined by a comparison of monitored vs. predicted control rod density.

2.0 DETAILED DESCRIPTION

The proposed change revises TS 3.1.2, "Reactivity Anomalies", as shown below.

LCO 3.1.2 The reactivity difference between the monitored ~~rod density~~ ***core k_{eff}*** and the predicted ~~rod density~~ ***core k_{eff}*** shall be within ± 1% Δk/k.

SR 3.1.2.1 Verify core reactivity difference between the monitored ~~rod density~~ ***core k_{eff}*** and the predicted ~~rod density~~ ***core k_{eff}*** is within ± 1% Δk/k.

The purpose of the reactivity anomaly surveillance is to compare the observed reactivity behavior of the core (at hot operating conditions) with the predicted reactivity behavior. Currently, DAEC TS 3.1.2 requires that the surveillance be done by comparing the monitored control rod density to the predicted control rod density, calculated prior to the start of operation for a particular cycle. The proposed amendment will change the method by which the reactivity anomaly surveillance is performed but not the specified frequency for performing the surveillance.

Current TS 3.1.2 requires that the reactivity equivalence of the difference between the monitored rod density and the predicted rod density shall not exceed ±1% Δk/k. The proposed amendment would revise TS 3.1.2 to state that the reactivity difference between the monitored core $k_{effective}$ (k_{eff}) and the predicted core k_{eff} shall not exceed ±1% Δk/k.

The current method of performing the reactivity anomaly surveillance uses rod density for the comparison primarily because early core monitoring systems did not calculate core critical k_{eff} values for comparison to design values. Instead, rod density was used as a convenient representation of core reactivity. Allowing the use of a direct comparison of core k_{eff}, as opposed to rod density, provides for a more direct measurement of core reactivity conditions and eliminates the limitations that exist for performing the core reactivity comparisons with rod density.

3.0 TECHNICAL EVALUATION

If a significant deviation between the reactivity observed during operation and the expected reactivity occurs, the reactivity anomaly surveillance alerts the reactor operating staff to a potentially anomalous situation, indicating that something in the core design process, the manufacturing of the fuel, or in the plant operation may be different than assumed. This situation would trigger an investigation and further actions as needed.

The current method for the development of the reactivity anomaly curves used to perform the TS surveillance actually begins with the predicted core k_{eff} at rated conditions and the companion rod patterns derived using those predicted values of k_{eff}. A calculation is made of

the number of notches inserted in the rod patterns, and also the number of equivalent notches required to make a change of ±1% reactivity around the predicted core k_{eff}. The rod density is converted to notches and plotted with an upper and lower bound representing the ±1% reactivity acceptance band as a function of cycle exposure. This curve is then used as the predicted rod density during the cycle. In effect, the comparison is indirect to critical core k_{eff} with a translation of acceptance criteria to rod density.

The revised method for evaluating a potential reactivity anomaly proposed in this amendment compares monitored core k_{eff} to predicted core k_{eff}. Monitored core k_{eff} is calculated by the 3D core simulator model in the plant's core monitoring system based on measured plant operating data. The predicted core k_{eff}, as a function of cycle exposure, is developed prior to the start of each operating cycle and incorporates benchmarking of exposure-dependent 3D core simulator k_{eff} behavior in previous cycles and any adjustments due to planned changes in fuel design, core design, or operating strategy for the upcoming cycle.

While being a convenient measurement of core reactivity, control rod density has its limitations, most obviously that all control rod insertion does not have the same impact on core reactivity. For example, edge rods and shallow rods (inserted 1/3 of the way into the core or less) have very little impact on reactivity while deeply inserted central control rods have a larger effect. Thus, a potential exists for reactivity anomaly concerns to arise during operations simply because of greater than anticipated use of near-edge and shallow control rods when in fact no true anomaly exists. Use of monitored to predicted core k_{eff} instead of rod density eliminates the limitations described above, provides for a technically superior comparison, and is a very simple and straightforward approach.

The proposed change will not affect transient and accident analyses because only the method of performing the reactivity anomaly surveillance is changing, and the proposed method will provide a technically superior comparison as discussed above. Furthermore, the reactivity anomaly surveillance will continue to be performed at the current required frequency. Consequently, core reactivity assumptions made in safety analyses will continue to be adequately verified, and no margins of safety will be reduced.

Regarding the core monitoring system, DAEC recently transitioned from the 3D MONICORE to the ACUMEN system, both from Global Nuclear Fuel (GNF). Reference 1 notes NRC acceptance of 3D MONICORE core surveillance system power distribution uncertainties, whereas Reference 2 notes acceptance for transition to ACUMEN. The latest version of the 3D MONICORE and ACUMEN systems incorporates the PANACEA Version 11 (PANAC11) core simulator code to calculate parameters such as core nodal powers, fuel thermal limits, etc., using actual, measured plant input data. Reference 3 notes NRC acceptance of PANAC11. PANAC11 is the same 3D core simulator code used in core design and licensing activities for DAEC. When a 3D MONICORE or ACUMEN core monitoring case is run, the core k_{eff} (as computed by PANAC11) is also calculated and printed directly on each 3D MONICORE or ACUMEN case output. This monitored value can then be directly compared to the predicted value of core k_{eff} as a measure of reactivity anomaly. No plant hardware or operational changes are required with this proposed change.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

General Design Criteria 26, 28, and 29 require that reactivity be controllable such that subcriticality is maintained under cold conditions and specified applicable fuel design limits are not exceeded during normal operations and anticipated operational occurrences. The reactivity anomaly surveillance required by the DAEC TS serves to partly satisfy the above General Design Criteria by verifying that core reactivity remains within expected/predicted values. Ensuring that no reactivity anomaly exists provides confidence of adequate shutdown margin as well as providing verification that the assumptions of safety analyses associated with core reactivity remain valid.

4.2 Precedent

The NRC has approved similar license amendments for the plants below that changed the method of performing the reactivity anomaly surveillance to use a comparison of monitored to predicted core k_{eff}.

- Peach Bottom - Amendments 284 and 287 [Reference 4]
- Hatch - Amendments 207 and 263 [Reference 5]
- Limerick - Amendments 168 and 207 [Reference 6]
- Brunswick - Amendments 187 and 218 [Reference 7]

In addition to the above amendments, reactivity anomaly surveillance requirement 3.1.2.1 in NUREG-1434, Standard Technical Specifications - General Electric BWR/6 Plants, Revision. 4.0, is written with the core k_{eff} comparison, as opposed to the control rod density comparison.

There are multiple BWRs that already use the core k_{eff} as the basis for the reactivity anomaly comparison. Reference 8 notes that the Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2 TS use the core k_{eff} comparison.

4.3 No Significant Hazards Consideration

The Duane Arnold Technical Specifications (TS) currently require performing the reactivity anomaly surveillance by comparing the monitored control rod density to the predicted control rod density calculated prior to the start of operation for a particular cycle. The proposed amendment will change the method by which the reactivity anomaly surveillance is performed by comparing monitored to predicted core reactivity.

As required by 10 CFR 50.91(a), NextEra has evaluated the proposed change to the Duane Arnold TS using the criteria in 10 CFR 50.92 and determined that the proposed change does not involve a significant hazards consideration. An analysis of the issue of no significant hazards consideration is presented below.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

The proposed change does not affect any plant systems, structures, or components designed for the prevention or mitigation of previously evaluated accidents. The proposed change would only modify how the reactivity anomaly surveillance is performed. Verifying that the core reactivity is consistent with predicted values ensures that accident and transient safety analyses remain valid. This amendment changes the TS requirements such that, rather than performing the surveillance by comparing monitored to predicted control rod density, the surveillance is performed by a direct comparison of core k_{eff}. Present day on-line core monitoring systems, such as 3D MONICORE and ACUMEN, are capable of performing the direct measurement of reactivity.

Therefore, since the reactivity anomaly surveillance will continue to be performed by a viable method, the proposed change does not involve a significant increase in the probability or consequence of a previously evaluated accident

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not involve any changes to the operation, testing, or maintenance of any safety-related, or otherwise important to safety systems. All systems important to safety will continue to be operated and maintained within their design bases. The proposed changes to the Reactivity Anomalies TS will only provide a new, more efficient method of detecting an unexpected change in core reactivity.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change is to modify the method for performing the reactivity anomaly surveillance from a comparison of monitored to predicted control rod density to a comparison of monitored to predicted core k_{eff}. The direct comparison of k_{eff} provides a technically superior method of calculating any differences in the expected core reactivity. The reactivity anomaly surveillance will continue to be performed at the same frequency as is currently required by the TS, only the method of performing the surveillance will be changed. Consequently, core reactivity assumptions made in safety analyses will continue to be adequately verified. The proposed change has no impact to the margin of safety.

Based on the above, NextEra concludes that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATIONS

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Letter from Frank Akstulewicz (NRC) to G. A. Watford (General Electric Company), "Acceptance for Referencing of Licensing Topical Reports NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations; NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluation; and Amendment 25 to NEDE-24011-P-A on Cycle-Specific Safety Limit MCPR (TAC Nos. M97490, M99069, and M97491," March 11, 1999.
2. Letter from Kevin Hsueh (NRC) to Jerald G. Head (GE-Hitachi Nuclear Energy Americas), "Final Safety Evaluation for Amendment 42 to Global Nuclear Fuel – Americas Topical Report NEDE-24011-P-A-US General Electric Standard Application for Reactor Fuel (GESTAR II) Supporting the Transition from the 3D-MONICORE Core Monitoring System to ACUMEN (CAC No. MF7438)," August 31, 2016.
3. Letter from Stuart A. Richards (NRC) to G. A. Watford (GE Nuclear Energy), "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, "GESTAR II" - Implementing Improved GE Steady-State Methods (TAC No. MA6481)," November 10, 1999.
4. Letter from Richard B. Ennis (U.S. Nuclear Regulatory Commission) to M. J. Pacilio (Exelon Nuclear), "Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments RE: Reactivity Anomalies Surveillance (TAC Nos. ME6356 and ME6357)," dated May 25, 2012.
5. Letter from Robert E. Martin (U.S. Nuclear Regulatory Commission) to M. J. Ajluni (Southern Nuclear Operating Company), "Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Issuance of Amendments Regarding Revision to Technical Specifications Limiting Condition for Operation 3.1.2, "Reactivity Anomalies" (TAC Nos. ME3006 and ME3007)," dated November 4, 2010.

6. Letter from Peter Bamford (U.S. Nuclear Regulatory Commission) to M. J. Pacilio (Exelon Nuclear), "Limerick Generating Station, Units 1 and 2 – Issuance of Amendments RE: Reactivity Anomalies Surveillance (TAC Nos. ME6348 and ME6349)," dated March 14, 2012.

7. Letter from David C. Trimble (U.S. Nuclear Regulatory Commission) to C. S. Hinnant (Carolina Power & Light Company), "Issuance of Amendment No. 187 to Facility Operating License No. DPR-71 and Amendment No. 218 to Facility Operating License No. DPR-62 Regarding a Change in the Methodology for Detecting a Reactivity Anomaly – Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. M97688 and M97689)," dated September 5, 1997.

8. Letter from M. D. Jesse (Exelon Generation Company) to U.S. Nuclear Regulatory Commission, "License Amendment Request - Reactivity Anomalies Surveillance," dated June 2, 2011.

ATTACHMENT 1

MARKUP OF TECHNICAL SPECIFICATIONS

2 pages follow

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

core keff

LCO 3.1.2 The reactivity difference between the monitored ~~rod density~~ and the predicted ~~rod density~~ shall be within ± 1% Δk/k.

core keff

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity difference not within limit.	A.1 Restore core reactivity difference to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the monitored ~~rod density~~ core keff and the predicted ~~rod density~~ core keff is within ± 1% Δk/k.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement AND 1000 MWD/T thereafter during operations in MODE 1

ATTACHMENT 2

MARKUP OF TECHNICAL SPECIFICATIONS BASES

5 pages follow

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with the UFSAR (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

(i.e., monitored)

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, and whatever neutron poisons (mainly xenon and samarium) that are

(continued)

BASES

core keff

BACKGROUND (continued)

present in the fuel. The predicted core reactivity, as represented by ~~control rod density~~, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. ~~The core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.~~

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The monitored core keff is calculated by the core monitoring system at actual plant conditions and is compared to the predicted value at the same cycle exposure.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted ~~rod density~~ core keff for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict ~~rod density~~ core keff may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured ~~rod density~~ core keff from the predicted ~~rod density~~ core keff that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Reactivity anomalies satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted core keff ~~rod density~~ of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, reactivity anomaly is not required during these conditions.

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally

(continued)

BASES

ACTIONS

A.1 (continued)

reviews the core conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the monitored and predicted ~~rod density~~ core keff is within the limits of the LCO provides added assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The Plant Process Computer calculates the ~~rod density~~ core keff for the reactor conditions obtained from plant instrumentation. A comparison of the monitored ~~rod density~~ core keff to the predicted ~~rod density~~ core keff at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

subsequent replacement with a new control rod or a control rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted ~~rod density~~ core keff can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. Additionally, the Reactor Engineer or individual fulfilling this role will normally be involved with determining equilibrium xenon conditions. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

REFERENCES

1. UFSAR, Sections 3.1.2.3.7, 3.1.2.3.9, and 3.1.2.3.10.
2. UFSAR, Chapter 15.