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10 CFR 50.90

June 2, 2011

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

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: License Amendment Request – Reactivity Anomalies Surveillance

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (Exelon) requests a proposed change to modify the Technical Specifications (TSs) concerning a change to the method of calculating core reactivity for the purpose of performing the reactivity anomaly surveillance at Peach Bottom Atomic Power Station, Units 2 and 3.

The proposed changes have been reviewed by the Peach Bottom Atomic Power Station Plant Operations Review Committee, and approved by the Nuclear Safety Review Board in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed amendment by June 2, 2012. Once approved, this amendment shall be implemented within 60 days of issuance. Additionally, there are no commitments contained within this letter.

Attachment 1 contains the evaluation of the proposed changes. Attachment 2 provides the marked up TS and Bases pages. The Bases pages are being provided for information only.

In accordance with 10 CFR 50.91, Exelon is notifying the State of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Officials.

Should you have any questions concerning this letter, please contact Tom Loomis at (610) 765-5510.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 2nd day of June 2011.

Respectfully,

Michael D. Jesse
Manager, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachment 1: Evaluation of Proposed Changes
Attachment 2: Markup of Technical Specifications and Bases Pages

cc: USNRC Region I, Regional Administrator
USNRC Senior Resident Inspector, PBAPS
USNRC Project Manager, PBAPS
R. R. Janati, Commonwealth of Pennsylvania

ATTACHMENT 1

EVALUATION OF PROPOSED CHANGES

ATTACHMENT 1

EVALUATION OF PROPOSED CHANGES

CONTENTS

SUBJECT: Reactivity Anomalies Surveillance

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Facility Operating Licenses DPR-44 and DPR-56 for Peach Bottom Atomic Power Station, Units 2 and 3.

The proposed change would revise the Technical Specifications to allow performance of the surveillance on a comparison of predicted to actual (or monitored) core reactivity. The reactivity anomaly verification is currently determined by a comparison of predicted vs. actual control rod density.

2.0 DETAILED DESCRIPTION

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, the PBAPS Technical Specifications (TSs) require that the surveillance be done by comparing predicted control rod density (calculated prior to the start of operation for a particular cycle) to an actual control rod density. The comparison is done, as required by the surveillance requirements. The proposed revision will change the method by which the reactivity anomaly surveillance is performed and not the specified frequency for performing the surveillance.

The current TS require that the reactivity equivalence of the difference between the actual rod density and the predicted rod density shall not exceed $\pm 1\%$ $\Delta k/k$.

The proposed TS and Bases would be revised to state that the reactivity difference between the actual $k_{effective}$ (k_{eff}) and the predicted k_{eff} shall not exceed $\pm 1\%$ $\Delta k/k$.

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 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.

Marked up TS Bases pages are provided in Attachment 2, and are provided for information only.

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 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 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 k_{eff} with a "translation" of acceptance criteria to rod density.

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, it is not uncommon for reactivity anomaly concerns to arise during operations simply because of greater use of near-edge and shallow control rods than anticipated, when in fact no true anomaly exists. Use of actual to predicted 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.

These proposed changes 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.

The following additional information is being provided concerning the core monitoring software as discussed in the Reference 1 request for additional information for the Edwin I. Hatch Nuclear Plant.

PBAPS utilizes the Global Nuclear Fuel (GNF) 3D MONICORE (Reference 2) core monitoring software system. The latest version of this product incorporates the PANACEA Version 11 (PANAC11) (Reference 3) core simulator code to calculate parameters such as core nodal powers, fuel thermal limits, etc., using actual, measured plant input data. PANAC11 is the same 3D core simulator code used in core design and licensing activities. When a 3D MONICORE core monitoring case is run, the core k_{eff} (as computed by PANAC11) is also calculated and printed directly on each 3D MONICORE case output. This value can then be directly compared to the predicted value of 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 PBAPS, Units 2 and 3 Technical Specifications 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 Precedents

A similar TS amendment was approved for:

1) Letter from R. E. Martin (U.S. Nuclear Regulatory Commission) to M. J. Ajluni (Southern Nuclear Operating Company, Inc), "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.

In addition to the above amendments, the reactivity anomaly Limiting Condition for Operation (LCO) in the BWR/6 Standard Technical Specifications, NUREG-1434, Rev. 3.0, is written with the k_{eff} comparison, as opposed to the control rod density comparison. The PBAPS, Units 2 and 3 TS comply with NUREG-1433, Revision 1, dated April 1995. In this revision of NUREG-1433, the comparison is a plant specific value in brackets.

Currently, 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 k_{eff} comparison.

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (Exelon) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

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

   Response: No.

   The proposed Technical Specifications changes do not affect any plant systems, structures, or components designed for the prevention or mitigation of previously evaluated accidents. The amendment would only change 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 Technical Specification requirements such that, rather than performing the surveillance by comparing predicted to actual control rod density, the surveillance is performed by a direct comparison of k_{eff}. Present day on-line core monitoring systems, such as the one in use at Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3 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 amendment does not involve a significant increase in the probability or consequence of a previously evaluated accident.

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

   Response: No.

   This Technical Specifications amendment request 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 anomaly Technical Specifications will only provide a new, more efficient method of detecting an unexpected change in core reactivity.

   Since all systems continue to be operated within their design bases, no new failure modes are introduced and the possibility of a new or different kind of accident is not created.

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

   Response: No.

   This proposed Technical Specifications amendment proposes to change the method for performing the reactivity anomaly surveillance from a comparison of predicted to actual control rod density to a comparison of predicted to actual 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 Technical Specifications, only the method of performing the surveillance will be changed. Consequently, core reactivity assumptions made in safety analyses will continue to be adequately verified.

   Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, Exelon Generation Company, LLC, concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.4 Conclusions

In conclusion, 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 CONSIDERATION

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 significant increase in the amounts of any effluent 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 M. J. Ajluni (Southern Nuclear Operating Company, Inc.) to U.S. Nuclear Regulatory Commission, "Information on 3D Monicore Core Monitoring Software," dated October 5, 2010.

2) MFN-003-99, F. Akstulewicz (NRC) to G. Watford (GE), Safety Evaluation Report for GE Licensing Topical Report NEDC-32694P, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations" (TAC No. M99069), March 11, 1999 [provides NRC acceptance of 3D MONICORE core surveillance system power distribution uncertainties].

3) MFN-035-99, S. Richards (NRC) to G. Watford (GE), "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 [provides NRC acceptance of PANACEA Version 11].

ATTACHMENT 2

Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3

Markup of Technical Specifications and Bases Pages

Revised Pages (Units 2 and 3)

3.1-5
3.1-6
B 3.1-8
B 3.1-9
B 3.1-10
B 3.1-11

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	core Keff Verify core reactivity difference between the monitored rod density and the predicted rod density is within ± 1% Δk/k. core Keff	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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

(ie., monitored)

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.

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 (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by

(continued)

BASES

core Keffective (Keff)

BACKGROUND (continued)

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.

The monitored core Keff is calculated by the core monitoring system 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 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 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 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 from the predicted rod density 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.

core Keff(s)

core Keff

core Keff

Reactivity anomalies satisfy Criterion 2 of the NRC Policy Statement.

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

(continued)

Core Keff

LCO (continued)

uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted ~~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. A deviation as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

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

<u>A.1</u>

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 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.

(continued)

BASES

ACTIONS

A.1 (continued)

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% $\Delta k/k$ 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

core Keff(s)

core Keff

core Keff

Verifying the reactivity difference between the monitored and predicted rod density 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 core monitoring system calculates the rod density for the reactor conditions obtained from plant instrumentation. A comparison of the monitored rod density to the predicted rod density 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 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 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

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

Core Kess

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

Core Kess

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	Core K_{eff} Verify core reactivity difference between the monitored ~~rod density~~ and the predicted ~~rod density~~ is within ± 1% $\Delta k/k$. Core K_{eff}	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement <u>AND</u> 1000 MWD/T thereafter during operations in MODE 1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

(i.e., monitored)

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.

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 (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by

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BASES

Core Keffective (Keff)

BACKGROUND (continued)

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

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

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.

Core Keff(s)

Core Keff

Core Keff

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 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 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 from the predicted rod density 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 the NRC Policy Statement.

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

(continued)

BASES

core Keff

LCO (continued)

uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted 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. A deviation as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

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 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.

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BASES

ACTIONS

A.1 (continued)

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

Core Keff(s)

core Keff

Core Keff

Verifying the reactivity difference between the monitored and predicted rod density 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 core monitoring system calculates the rod density for the reactor conditions obtained from plant instrumentation. A comparison of the monitored rod density to the predicted rod density 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 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 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

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