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SOUTHERN COMPANY
Energy to Serve Your World℠

December 17, 2009

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

Edwin I. Hatch Nuclear Plant
Revision to Technical Specifications LCO 3.1.2, Reactivity Anomalies

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90 Southern Nuclear Operating Company (SNC) hereby requests a change to the Plant Hatch Units 1 and 2 Technical Specifications, Appendix A to Operating Licenses DPR-57 and NPF-5, respectively.

SNC requests a change to Limiting Condition for Operation (LCO) 3.1.2, Reactivity Anomalies. The amendment requests a change to the method of calculating core reactivity for the purpose of performing the anomaly check.

Enclosure 1 provides a description and justification of the proposed change, the significant hazards evaluation, and the justification for the exclusion from performing an environmental evaluation. Enclosure 2 and 3 provide the marked-up and clean typed Technical Specifications and Bases pages, respectively.

Approval of the proposed amendment is requested by December 17, 2010. Southern Nuclear Operating Company requests that the proposed licensing amendments be effective on the date of issuance and implementation required within 60 days following issuance.

This letter contains no NRC commitments. If you have any questions, please advise.

A001
NRR

Mr. M. J. Ajluni states he is Manager-Nuclear Licensing of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Manager Nuclear Licensing

Sworn to and subscribed before me this 17th *day of* December, *2009.*

Patricia H. Raymond
Notary Public

My commission expires: 7-21-2012

MJA/PAH/lac

Enclosures:
1. Basis for Proposed Change
2. Technical Specifications Markup Pages
3. Technical Specifications Clean Typed Pages

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Edwin I. Hatch Nuclear Plant
Revision of LCO 3.1.2, Reactivity Anomalies

Enclosure 1

Basis for Proposed Change

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1.0 Summary Description

This evaluation supports a request to amend Operating Licenses DPR-57 and NPF-5 of the Plant Hatch Units 1 and 2 Technical Specifications.

The proposed change would revise LCO 3.1.2, "Reactivity Anomalies" to allow performance of the surveillance on a comparison of predicted to measured (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 expected reactivity behavior calculated prior to the start of operation.

Currently, Hatch Technical Specifications (TS) require that the check be done by comparing a 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 Surveillance Requirement (SR) 3.1.2.1, once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement, and each 1000 MWD/T thereafter during operations in Mode 1. This proposed TS change will not change the frequency of the SR.

The proposed revision will change the method by which the reactivity anomaly surveillance is performed.

The current LCO 3.1.2 reads:

The reactivity difference between the actual rod density and the predicted rod density shall be within +/- 1 % $\Delta k/k$.

The proposed LCO 3.1.2 reads:

The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall be within +/- 1 % $\Delta k/k$.

The Conditions statement will not be changed.

The SR will also be re-worded to replace rod density with $k_{effective}$ (k_{eff}) as appropriate.

The current method of performing the reactivity anomaly 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. Rod density was used instead 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.

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 of 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 critical 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 average notches required to make a change of +/- 1 % $\Delta k/k$ around the predicted critical k_{eff}. The notches are converted to rod density and plotted with an upper and lower bound representing the +/- 1 % $\Delta k/k$ 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 still based on critical k_{eff} with a "translation" of acceptance criteria to rod density.

The revised method for evaluating a potential reactivity anomaly compares measured core K_{eff} to predicted core K_{eff}. Measured 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 fuel vendor recommended 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 have very little impact on reactivity while deeply inserted central control rods have a large effect. Thus, it is not uncommon for reactivity anomaly concerns to arise during operation simply because of greater use of near-edge or 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 an adequate estimate as discussed above. Furthermore, the anomaly check will continue to be performed at the current required frequency.

4.0 Regulatory Evaluation

4.1 Significant Hazards Consideration

This amendment proposes to change the reactivity anomaly limiting condition for operation (LCO 3.1.2, "Reactivity Anomalies") to allow a direct measurement of core reactivity by using the multiplication factor (k_{eff}) rather than the control rod density.

Southern Nuclear 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 described below:

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

   Response: No.

   This proposed Technical Specifications change does 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 check 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 LCO 3.1.2 and SR 3.1.2.1 requirements such that, rather than performing the check by comparing predicted to actual control rod density, the check is performed by a direct comparison of k_{eff}. Present day on-line core monitoring systems, such as the one in use at Plant Hatch, are capable of performing the direct measurement of reactivity.

   Therefore, since the reactivity anomaly check 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 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, system. All important to safety systems will continue to be operated, surveillances performed, and maintained within their design bases. The proposed changes to the reactivity anomaly LCO 3.1.2 and SR 3.1.2.1 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 LCO 3.1.2 and SR 3.1.2.1 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 check will continue to be performed at the same frequency as is currently required by the Tech Specs, only the method of performing the check will be changed. Consequently, core reactivity assumptions made in safety analyses will continue to be adequately verified.

   The proposed amendment does not therefore involve a significant reduction in a margin of safety.

4.2 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 check required by the Hatch Technical Specifications in LCO 3.1.2 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.

4.3 Precedent

This Technical Specifications amendment was granted to the Brunswick Steam Electric Plant on September 5, 1997 via Amendments 187 and 218 for Units 1 and 2. The Brunswick plant, like Plant Hatch, is a BWR/4 nuclear reactor plant. The Technical Specifications changes that Plant Hatch is requesting are the same as those granted to the Brunswick Plant in the referenced amendments.

Additionally, the Reactivity Anomaly LCO in the BWR/6 Standard Technical Specifications, NUREG-1434, Rev. 3.1, is written with the k_{eff} comparison, as opposed to the control rod density comparison. The Bases changes are modeled after the applicable BWR/6 Standard Bases.

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

There is no physical hardware change to any structure, system, or component within the plant. 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 amounts of any effluents that may be released off site, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria 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. Hatch Technical Specifications Bases, B 3.1.2, "Reactivity Anomalies."
2. 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). David C. Trimble, Project Manager, Office of Nuclear Reactor Regulation to C.S. Hinnant, Vice President Carolina Power and Light Company, Brunswick Steam Electric Plant, September 5, 1997.
3. NUREG 1434, Standard Technical Specifications — General Electric Plants (BWR/6)
4. General Design Criteria (GDC) 26, "Reactivity Control System Redundancy and Capability"; GDC 27, "Combined Reactivity Control System Capability"; GDC 28, "Reactivity Limits."

Edwin I. Hatch Nuclear Plant
Revision of LCO 3.1.2, Reactivity Anomalies

Enclosure 2

Technical Specifications and Bases Markup Pages

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

monitored core k_{eff}

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

core k_{eff}

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 ~~actual rod density~~ and the predicted ~~rod density~~ 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

monitored core k_{eff}

core k_{eff}

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

measured (i.e., monitored)

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and specified acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity anomaly is used as a measure of the predicted versus ~~actual~~ 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 ~~actual~~ 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.

measured

(

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measured

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 ~~actual~~ 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 poison, 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 poisons (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 ~~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~~

core $k_{effective}$ (k_{eff})

(continued)

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

BASES

BACKGROUND (continued)

~~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 ~~actual~~ and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the ~~actual~~ 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 ~~actual~~ and predicted core reactivity exists at BOC, then the prediction may be normalized to the ~~actual~~ value. Thereafter, any significant deviations in the ~~actual 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.

measured core k_{eff}

Reactivity anomalies satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

measured

measured

core $k_{eff(s)}$

core k_{eff}

measured

measured

core k_{eff}

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 ~~actual~~ 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.

monitored core k_{eff}

core k_{eff}

(continued)

BASES (continued)

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. 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

(continued)

BASES

ACTIONS

B.1 (continued)

full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

monitored

monitored core k_{eff}

core $k_{eff(s)}$

core monitoring system

core k_{eff}

Verifying the reactivity difference between the ~~actual~~ 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 ~~Process Computer~~ calculates the ~~rod density~~ for the reactor conditions obtained from plant instrumentation. A comparison of the ~~actual 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 flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T (short ton) 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.

core $k_{eff(s)}$

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. FSAR, Chapter 14.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

monitored core k_{eff}

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

core k_{eff}

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 ~~actual rod density~~ and the predicted ~~rod density~~ 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

monitored core k_{eff}

core k_{eff}

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

measured (i.e., monitored)

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and specified acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity anomaly is used as a measure of the predicted versus ~~actual~~ 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 ~~actual~~ 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.

measured

measured

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 ~~actual~~ 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 poison, 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 poisons (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel.

core $k_{effective}$ (k_{eff})

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~~

(continued)

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

BASES

BACKGROUND (continued)

~~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.

measured

measured

core $k_{eff(s)}$

The comparison between ~~actual~~ and predicted initial core reactivity provides a normalization for the ~~calculational~~ models used to predict core reactivity. If the ~~actual~~ 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 ~~actual~~ and predicted core reactivity exists at BOC, then the prediction may be normalized to the ~~actual~~ value. Thereafter, any significant deviations in the ~~actual 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 k_{eff}

measured

measured

core k_{eff}

measured core k_{eff}

Reactivity anomalies satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

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 ~~actual~~ 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.

monitored core k_{eff}

core k_{eff}

(continued)

BASES (continued)

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. 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

(continued)

BASES

ACTIONS

B.1 (continued)

full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

monitored

monitored core k_{eff}

core $k_{eff(s)}$

core monitoring system

core k_{eff}

Verifying the reactivity difference between the ~~actual~~ 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~~ ~~Process Computer~~ calculates the ~~rod density~~ for the reactor conditions obtained from plant instrumentation. A comparison of the ~~actual 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 flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T (short ton) 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.

core $k_{eff(s)}$

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. FSAR, Chapter 15.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

Edwin I. Hatch Nuclear Plant
Revision of LCO 3.1.2, Reactivity Anomalies

Enclosure 3

Technical Specifications and Bases Clean Typed Pages

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

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

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 core k_{eff} and the predicted core k_{eff} 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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and specified acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity anomaly is used as a measure of the predicted versus measured (i.e., monitored) 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 poison, 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 poisons (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 core $k_{effective}$ (k_{eff}), 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 monitored core k_{eff} is calculated by the core monitoring system for actual plant conditions and is then compared to the predicted value for the cycle exposure.

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BASES (continued)

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 core $k_{eff(s)}$ 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 core k_{eff} 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 core k_{eff} from the predicted core k_{eff} 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 (Ref. 3).

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 core k_{eff} and the predicted core k_{eff} of $\pm$ 1% $\Delta k/k$ 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

(continued)

BASES

APPLICABILITY (continued)

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. 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

Verifying the reactivity difference between the monitored and predicted core $k_{eff(s)}$ is within the limits of the LCO provides added assurance that plant operation is maintained within the assumptions

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

of the DBA and transient analyses. The core monitoring system calculates the core k_{eff} for the reactor conditions obtained from plant instrumentation. A comparison of the monitored core k_{eff} to the predicted core k_{eff} 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 core $k_{eff(s)}$ 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. The 1000 MWD/T (short ton) 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. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. FSAR, Chapter 14.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

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APPLICABILITY: MODES 1 and 2.

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B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

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SURVEILLANCE		FREQUENCY
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and specified acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity anomaly is used as a measure of the predicted versus measured (i.e., monitored) 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.

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BASES

APPLICABILITY (continued)

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SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

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