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

Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Request for License Amendment Regarding Effect of Rod Shadowing on Excore Power Range Detectors, Changes to Nuclear Overpower Reactor Trips, and Changes to Monitoring of Moderator Temperature Coefficient and Core Power Distribution (LAR-17-031)

Ladies and Gentlemen:

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to the combined licenses (COLs) for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 (License Numbers NPF-91 and NPF-92, respectively). The proposed amendment would revise the licensing basis information regarding the following:

- Design of the protection and safety monitoring system (PMS) automatic reactor trips and the crediting of PMS automatic reactor trips necessary to prevent exceeding the fuel design limits including the power range high neutron flux (high setpoint) trip, the power range high positive flux rate trip, the overpower ΔT trip, and the overtemperature ΔT trip, and

- Changes to the Combined License (COL) Appendix A Technical Specifications for maintaining moderator temperature coefficient (MTC) and maintaining power distributions within the required absolute power generation limits.

These changes are proposed to maintain compliance with the following 10 CFR Part 50, Appendix A, General Design Criteria:

- General Design Criterion (GDC) 10, which requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences,

- GDC 11, which requires that the reactor core and associated coolant systems be designed so that in the power operating range the net effect of the prompt inherent

nuclear feedback characteristics tends to compensate for a rapid increase in reactivity, and

- GDC 12, which requires a core design to assure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible.

The requested amendment proposes to depart from approved AP1000 Design Control Document (DCD) Tier 2 information (text, tables, and figures) as incorporated into the Updated Final Safety Analysis Report (UFSAR) as plant-specific DCD information, and also proposes to depart from involved plant-specific Technical Specifications (PS-TS) as incorporated in Appendix A of the COL. Associated PS-TS Bases document revisions that will be incorporated coincident with the requested amendment are provided for information.

Enclosure 1 provides the description, technical evaluation, regulatory evaluation (including the significant hazards consideration determination), and environmental considerations for the proposed changes.

Enclosure 2 provides markups depicting the requested changes to the VEGP Units 3 and 4 licensing basis documents.

Enclosure 3 provides the conforming changes to the Technical Specifications Bases for information only.

This letter, including enclosures, has been reviewed and confirmed to not contain security-related information. This letter contains no regulatory commitments.

SNC requests NRC staff review and approval of the license amendment no later than March 30, 2018. Approval by this date will allow sufficient time to implement licensing basis changes necessary to support procedure development in relation to conducting the necessary operator training to support plant operations. SNC expects to implement the proposed amendment (through incorporation into the licensing basis documents (e.g., the UFSAR) within 30 days of approval of the requested changes.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia by transmitting a copy of this letter and its enclosures to the designated State Official.

Should you have any questions, please contact Mr. Wesley Sparkman at (205) 992-5061.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 8th of September 2017.

Respectfully submitted,

Brian H. Whitley
Director, Regulatory Affairs
Southern Nuclear Operating Company

Enclosures

1) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 - Request for License Amendment Regarding Effect of Rod Shadowing on Excore Power Range Detectors, Changes to Nuclear Overpower Reactor Trips, and Changes to Monitoring of Moderator Temperature Coefficient and Core Power Distribution (LAR-17-031)

2) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Proposed Changes to the Licensing Basis Documents (LAR-17-031)

3) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 - Conforming Changes to the Technical Specifications Bases (For Information Only) (LAR-17-031)

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Southern Nuclear Operating Company

ND-17-1495

Enclosure 1

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Request for License Amendment Regarding
Effect of Rod Shadowing on Excore Power Range Detectors,
Changes to Nuclear Overpower Reactor Trips, and
Changes to Monitoring of Moderator Temperature Coefficient and Core Power Distribution

(LAR-17-031)

(This Enclosure consists of 53 pages, including this cover page)

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Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests an amendment to Combined License (COL) Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

1. SUMMARY DESCRIPTION

The proposed changes affect the design of the protection and safety monitoring system (PMS) automatic reactor trips and the crediting of PMS automatic reactor trips necessary to prevent exceeding the fuel design limits including the power range high neutron flux (high setpoint) trip, the power range high positive flux rate trip, the overpower ΔT trip, and the overtemperature ΔT trip, and changes to the COL Appendix A Technical Specifications for maintaining moderator temperature coefficient (MTC) and maintaining power distributions within the required absolute power generation limits. These activities include the following:

1. A revised form of axial flux difference (AFD) is proposed as input to the f(ΔI) penalty function generator, which provides input to the overpower ΔT and overtemperature ΔT trip setpoints in the PMS, to account for increased uncertainties in the power range detector indicated reactor power level during mechanical shim (MSHIM) operations.

2. The overpower ΔT trip, and for rapid power increases the power range high positive flux rate trip, replace the power range high neutron flux (high setpoint) trip to account for increased uncertainties in the power range detector indicated reactor power level during MSHIM operations.

3. COL Appendix A Technical Specifications 3.1.3, Moderator Temperature Coefficient (MTC), and 5.6.3, CORE OPERATING LIMITS REPORT (COLR), are revised to reflect changes in methodology for determining MTC at beginning of life (BOL) and end of life (EOL).

4. COL Appendix A Technical Specifications 3.2, Power Distribution, are revised to reflect the proposed changes to the design of the PMS automatic reactor trips above, and to incorporate various changes to improve consistency in the current licensing basis. This includes the following:

    a. COL Appendix A Technical Specification 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z)),

    b. COL Appendix A Technical Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and

    c. COL Appendix A Technical Specification 3.2.5, On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters.

5. COL Appendix A Technical Specification 3.3.1, Reactor Trip System (RTS) Instrumentation, is revised to reflect the proposed changes to the design of the PMS automatic reactor trips described above.

6. COL Appendix A Technical Specification 3.7.1, Main Steam Safety Valves (MSSVs), is revised to reflect the proposed changes to the design of the PMS automatic reactor trips described above.

The requested amendment proposes changes to the UFSAR in the form of departures from the plant-specific Design Control Document (DCD) Tier 2 information (as detailed in Section 2), and involves changes to COL Appendix A Technical Specifications. This enclosure requests approval of the license amendment necessary to implement the Tier 2 and COL Appendix A changes. Proposed changes to the Technical Specifications Bases are provided for information only.

2. DETAILED DESCRIPTION and TECHNICAL EVALUATION

2.1 PMS Power Range Detector Indicated Reactor Power Level and AFD Uncertainties Due to Control Rod Shadowing during MSHIM Operation

Calibration of the power range detector AFD inputs to the f(ΔI) penalty function generator, which provides input to the overpower ΔT and overtemperature ΔT trip setpoints in the PMS, does not appropriately account for increased uncertainties in the power range detector indicated reactor power level during MSHIM operations. The original AFD input uncertainty assumes that the AFD inputs from the power range detector channels are all within 3% of the true core average delta-flux (ΔΦ) value. The value of core average ΔΦ is a representation of the difference between the average thermal power in the top and bottom halves of the reactor core. The motion of the control rods during MSHIM operations changes the relationship between the power range detector measurements and the true value of ΔΦ. The changes in the calibration relationships during MSHIM operations is caused by the fact that essentially all of the neutron flux measured by the power range detectors comes from a very limited number of fuel assemblies on the periphery of the reactor core. The motion of the control rods changes the relationship between the core average ΔΦ and the ΔΦ that exists in the fuel assemblies on the core periphery that produce the neutrons seen by the power range detectors. This results in the core average ΔΦ input to the f(ΔI) penalty calculation exceeding the original error allowance. Therefore, to avoid this increased uncertainty, a more accurately measurable form of ΔΦ is proposed to be used as input to the f(ΔI) penalty function generator using weighted peripheral axial (WPA) ΔΦ rather than core average ΔΦ.

Since neutron flux that is seen by the power range detectors comes primarily from peripheral assemblies, a very stable and accurate calibration between the power range detector signals and a WPA ΔΦ can be developed. This allows the WPA ΔΦ to replace the core average ΔΦ values currently used to calculate the amount of f(ΔI) penalty used as input to the overpower ΔT and overtemperature ΔT trip setpoints calculations. Implementation of this approach produces an f(ΔI) penalty function relationship that has a significantly different shape than the typical relationships seen in most Westinghouse PWR designs. However, a detailed analysis demonstrates the acceptability of implementing this approach relative to core design, nuclear safety analysis, and reactor trip setpoint generation requirements. The analysis also demonstrates that there is no loss of operating margin, although there is some reduction of the existing excess axial power distribution

related core design margin in the AP1000 plant. However, adequate margin remains in the core design as demonstrated in the revised Three-Dimensional Final Acceptance Criteria (3D FAC) power distribution analysis for AP1000 plant Cycle 1 cores. The 3D FAC power distribution analysis confirms several COL Appendix A Technical Specifications limits when the OPDMS is inoperable, generates OPDMS inoperable COLR surveillance W(Z) functions, and confirms fuel integrity limits are met during several Condition II accidents initiating from both OPDMS operable and inoperable conditions.

This activity affects the design of the power range nuclear detectors, including the increased uncertainties in the power range detector indicated reactor power level during MSHIM operations, and impact on the measurement of the axial heat flux distribution.

Control and Monitoring of Core Power Stability during Normal Operations

As described in Updated Final Safety Analysis Report (UFSAR) (plant-specific) DCD Subsection 4.3.2.2.4, the distribution of power in the axial or vertical direction is largely under the control of the operator through either the manual operation of the control rods or the automatic motion of control rods in conjunction with manual operation of the chemical and volume control system. The automated mode of operation is referred to as MSHIM and is discussed in UFSAR Subsection 4.3.2.4.16. The rod control system has an automatic and manual mode of operation. In the automatic mode, the rod control system automatically modulates the insertion of the axial offset (AO) control bank controlling the axial power distribution simultaneous with the MSHIM gray and control rod banks to maintain programmed coolant temperature. Operation of the chemical and volume control system is initiated manually by the operator to compensate for fuel burnup and maintain the desired MSHIM bank insertion (typically every two weeks). Nuclear effects which cause variations in the axial power shape include moderator density, Doppler effect on resonance absorption, spatial distribution of xenon, burnup, and axial distribution of fuel enrichment and burnable absorber. Automatically controlled variations in total power output and rod motion are also important in determining the axial power shape at any time.

As described in UFSAR Subsection 4.3.2.7.6.1, Axial Power Distribution, the rod control system automatically maintains axial power distribution within very tight AO bands as part of normal operation when in the automatic mode. The AO control bank is specifically designed with sufficient worth to be capable of maintaining essentially constant axial offset over the power operating range. The rod control system is also allowed to be operated in manual control in which case the operator is instructed to maintain an axial offset within a prescribed operating band, based on the excore detector readings. Should the axial offset be permitted to move far enough outside this band, the protection limit is encroached, and the turbine power is automatically reduced or a reactor trip signal generated, or both.

As fuel burnup progresses, pressurized water reactor (PWR) cores become less stable to axial xenon oscillations. However, free xenon oscillations are not allowed to occur, except for special tests. The AO control bank is sufficient to dampen and control any axial xenon oscillations present. Should the AO be inadvertently permitted to move far enough outside the allowed band due to an axial xenon oscillation or for any other reason, the overpower ΔT and/or overtemperature ΔT trip setpoints, including the AO compensation,

are reached and the turbine power is automatically reduced and a reactor trip signal is generated.

As described in UFSAR Subsection 4.4.4.3.2, Axial Heat Flux Distributions, the axial heat flux distribution can vary as a result of rod motion or power change or as a result of a spatial xenon transient which may occur in the axial direction. The excore nuclear detectors, as described in UFSAR Subsection 4.3.2.2.7, are used to measure the axial power imbalance. The information from the excore detectors is used to protect the core from excessive axial power imbalance. The reference axial shape used in establishing core departure from nucleate boiling (DNB) limits (i.e., the overtemperature ΔT trip setpoint) is a chopped cosine with a peak-to-average value of 1.61. The reactor protection system provides automatic reduction of the trip setpoint on excessive axial power imbalance. To determine the magnitude of the setpoint reduction, the reference shape is supplemented by other axial shapes skewed to the bottom and top of the core.

The course of those accidents in which DNB is a concern is analyzed in UFSAR Chapter 15 assuming that the protection setpoints have been set on the basis of these shapes. In many cases, the axial power distribution in the hot channel changes throughout the course of the accident due to rod motion, coolant temperature, and power level changes.

The initial conditions for the accidents for which DNB protection is required are assumed to be those permissible within the specified axial offset control limits described in UFSAR Subsection 4.3.2.2. In the case of the loss-of-flow accident, the hot channel heat flux profile is very similar to the power density profile in normal operation preceding the accident. It is therefore possible to illustrate the calculated minimum DNB ratio (DNBR) for conditions representative of the loss-of-flow accident as a function of the flux difference initially in the core. The power shapes are evaluated with an $F^N_{\Delta H}$ of 1.654 (= 1.72 / 1.04). The radial contribution to the hot rod power shape is conservative both for the initial condition and for the condition at the time of minimum DNBR during the loss-of-flow transient. The minimum DNBR is calculated for the design power shape for non-overpower/overtemperature DNB events. This design shape results in calculated DNBR that bounds the normal operation shapes.

The OPDMS provides the operator with detailed power distribution information in both the radial and axial sense continuously using signals from the fixed incore detectors. Signals are also available to the operator from the excore ion chambers (i.e., the power range detector channels), which are long ion chambers outside the reactor vessel running parallel to the axis of the core. Separate signals are taken from each ion chamber. The ion chamber signals are processed and calibrated against incore measurements such that an indication of the power in the top of the core less the power in the bottom of the core is derived. The calibrated difference in power between the core top and bottom halves, called the flux difference (ΔI), is derived for each of the four channels of excore detectors and is displayed on the control panel. The principal use of the flux difference is to provide the shape penalty function (f(ΔI)) to the overpower ΔT trip for the overpower core protection function and the overtemperature ΔT trip for the DNB core protection function.

Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.3.1

To address this issue, COL Appendix A Technical Specification 3.3.1 is proposed to be revised as follows:

- Note 1 modifying SR 3.3.1.2 is proposed to be deleted, and item a under Note 3 (changed to Note 2) modifying SR 3.3.1.2 is proposed to be revised to change the criteria for when the calorimetric heat balance to nuclear instrument channel output is required to be compared from "if the nuclear instrumentation channel indicated power is lower than the calorimetric measurement by > 1%" to "if the nuclear instrumentation channel indicated power is lower than the calorimetric measurement by > 5% RTP [Rated Thermal Power]" as a less restrictive change. In addition, Note 1 modifying SR 3.3.1.3 is proposed to be revised to change the criteria for when the calorimetric heat balance to the ΔT power calculation ($q_{\Delta T}$) output is required to be compared from "if absolute difference between $q_{\Delta T}$ [from the reactor coolant system side] and the calorimetric measurement [secondary power side] is > 1% RTP" to "if absolute difference between $q_{\Delta T}$ and the calorimetric measurement is > 3% RTP" as a less restrictive change. These proposed changes are included in the setpoint calculations that demonstrate recovery of setpoint margin to obtain acceptable results for the overpower ΔT and overtemperature ΔT trip functions as described in Section 2.3. As previously described, a detailed analysis demonstrates the acceptability of implementing this approach relative to core design, nuclear safety analysis, and reactor trip setpoint generation requirements. The analysis also demonstrates there is no loss of operating margin, although there is some reduction of the existing excess axial power distribution related core design margin in the AP1000 plant. However, adequate margin remains in the core design. The UFSAR Chapter 15 safety analyses are not impacted by this change to these Notes modifying SR 3.3.1.2, because the change does not affect any of the inputs, methodology, assumptions, or acceptance criteria that are modeled in the safety analyses. No changes to the UFSAR Chapter 15 safety analyses result from this change. Therefore, these proposed changes are acceptable to prevent exceeding fuel design limits during design basis events in MODE 1 with THERMAL POWER ≥ 15% RTP.

- Note 3 (changed to Note 2) modifying SR 3.3.1.2 is proposed to be revised to change the criteria for when the calorimetric heat balance to nuclear instrument channel output is required to be compared from "If the calorimetric heat balance is < 70% RTP" to "If the calorimetric heat balance is ≥ 15% RTP" as a more restrictive change. These proposed changes are acceptable, as the proposed changes address the increased uncertainties due to control rod shadowing during MSHIM operation to calculate an acceptable f(ΔI) penalty used as input to the overpower ΔT and overtemperature ΔT trip setpoints at low and mid power levels. The UFSAR Chapter 15 safety analyses are not impacted by this change to this Note modifying SR 3.3.1.2, because the change does not affect any of the inputs, methodology, assumptions, or acceptance criteria that are modeled in the safety analyses. No changes to the UFSAR Chapter 15 safety analyses result from this change. Therefore, these proposed changes are acceptable to prevent

exceeding fuel design limits during design basis events in MODE 1 with THERMAL POWER ≥ 15% RTP.

Description of any Changes to Current Licensing Basis Documents

COL Appendix A Technical Specifications Changes:

The following changes to the COL Appendix A Technical Specifications are proposed:

1. Technical Specification 3.3.1 is revised as follows:

   a. SURVEILLANCE REQUIREMENT SR 3.3.1.2 is revised to delete the first NOTE that states: "Adjust nuclear instrument channel in the Protection and Safety Monitoring System (PMS) if absolute difference is > 1% RTP." and to renumber the subsequent notes.

   b. SURVEILLANCE REQUIREMENT SR 3.3.1.2 NOTE 3, renumbered as NOTE 2, is revised to replace "< 70% RTP" with "≥ 15% RTP" for when the note is applicable. In addition, NOTE 3.a, renumbered as NOTE 2.a, is revised to replace "> 1%" with "> 5% RTP" for the difference between the calorimetric heat balance and nuclear instrumentation channel indicated power at which time the nuclear instrumentation channel is required to be adjusted upward to match the calorimetric measurement.

   c. SURVEILLANCE REQUIREMENT SR 3.3.1.3 NOTE 1 is revised to replace "> 1% RTP" with "> 3% RTP" for the absolute difference between $q_{\Delta T}$ and the calorimetric measurement at which time the conversion factor, ΔT°, in the ΔT power calculation ($q_{\Delta T}$), is required to be adjusted.

UFSAR Changes:

The following licensing basis changes to the UFSAR are proposed:

None.

2.2 Use of Overpower ΔT Trip and Power Range High Positive Flux Rate Trip for Reactor Protection During MSHIM Operation

Because of the current uncertainties in power range detector indicated reactor power level, the error allowances for the power range high neutron flux (high setpoint) trip are exceeded, and the current power range high neutron flux (high setpoint) trip does not provide adequate reactor protection so that the acceptance limit of 118% RTP is not exceeded. Therefore, the overpower ΔT trip, and for rapid power increases the power range high positive flux rate trip, replace the power range high neutron flux (high setpoint) trip to account for increased uncertainties in the power range detector indicated reactor power level during MSHIM operations.

COL Appendix A Technical Specification 3.2.1 requires changes to address this issue as follows:

- Required Action A.2 requires the operator to "Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit" for the Condition of $F_Q^C(Z)$ not within limit as specified in the COLR. In addition, Required Action B.2 requires the operator to "Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit" for the Condition of $F_Q^W(Z)$ not within limits as specified in the COLR. Because the overpower ΔT trip, and for rapid power increases the power range high positive flux rate trip, replace the power range high neutron flux (high setpoint) trip to account for increased uncertainties in the power range detector indicated reactor power level during MSHIM operations, the reduction of the power range high neutron flux (high setpoint) trip setpoints are not necessary to protect fuel design limits.

COL Appendix A Technical Specification 3.2.2 requires changes to address this issue as follows:

- Required Action A.1.2.2 requires the operator to "Reduce Power Range Neutron Flux – High trip setpoints to ≤ 55%" for the Condition of $F^N_{\Delta H}$ not within limits as specified in the COLR. Because the overpower ΔT trip, and for rapid power increases the power range high positive flux rate trip, replace the power range high neutron flux (high setpoint) trip to account for increased uncertainties in the power range detector indicated reactor power level during MSHIM operations, the reduction of the power range high neutron flux (high setpoint) trip setpoints are not necessary to protect fuel design limits. However, the overpower ΔT trip setpoint is required to be reduced instead to protect fuel design limits when $F^N_{\Delta H}$ is not within limits as specified in the COLR.

COL Appendix A Technical Specification 3.7.1 requires changes to address this issue as follows:

- Required Action A.2 requires the operator to "Reduce the Power Range Neutron Flux – High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs." for the Condition of one or both steam generators with one or more MSSVs inoperable for opening. Because the overpower ΔT trip replaces the power range high neutron flux (high setpoint) trip to account for increased uncertainties in the power range detector indicated reactor power level during MSHIM operations for the limiting event occurring in MODE 1 that may result in secondary system overpressurization (i.e., a rod cluster control assembly (RCCA) bank withdrawal at power event), the reduction of the overpower ΔT trip setpoints is more appropriate.

For events occurring at 100% RTP, the power range high positive flux rate trip (with a presumed trip setpoint of 15% increase and a 60-second time constant) provides margin to the 118% RTP acceptance limit, and even larger margins for events occurring at less than full power. In addition, the power range high positive flux rate trip setpoint is self-calibrating in the sense that it continually adjusts its reference value (the lagged signal value) toward the current indicated neutron flux. For the power range high neutron flux (high setpoint) trip, instrument drift and process measurement errors impact the calibration of the reactor trip

setpoint. For events that are not rapid, the overpower ΔT trip provides margin to the 118% RTP acceptance limit. The overlap between the responses of the power range high positive flux rate trip and overpower ΔT trip during design basis events is detailed in the updates to the UFSAR Chapter 15 safety analyses.

As a result of the increased uncertainties in the power range detector indicated reactor power level during MSHIM operations, the UFSAR Chapter 15 safety analyses no longer credit the power range high neutron flux (high setpoint) reactor trip function. Instead, overpower protection is provided by a combination of the power range high positive flux rate, the overpower ΔT, and the overtemperature ΔT reactor trip functions. Therefore, UFSAR markups are provided throughout UFSAR Chapter 15 to indicate that, although the power range high neutron flux (high setpoint) remains available for protection for overpower transients, the function is no longer credited in any of the safety analyses for primary protection. The only UFSAR Chapter 15 transient that is reanalyzed to incorporate this change is the uncontrolled RCCA bank withdrawal at power event analysis (UFSAR Section 15.4.2). The revised detailed analysis, which models the combination of the power range high positive flux rate, the overpower ΔT, and the overtemperature ΔT reactor trip functions, continues to demonstrate that the DNB design basis is met. The spectrum of RCCA ejection accidents analysis (UFSAR Section 15.4.8) is the only other UFSAR Chapter 15 safety analysis that credits the power range high neutron flux (high setpoint) reactor trip function. However, that analysis already reflects primary protection also provided by the power range high positive flux rate trip. As such, no new detailed analysis of the spectrum of RCCA ejection accidents transients was required to incorporate this change. The resulting impacts and required changes to address this issue are further described below.

As described in UFSAR Section 15.1, Increase in Heat Removal From the Primary System, a number of events that could result in an increase in heat removal from the reactor coolant system are postulated. Detailed analyses are evaluated for the events that have been identified as limiting cases to include the proposed changes to the power range high positive flux rate trip, and the overpower ΔT and overtemperature ΔT reactor trips, including the following:

- As described in UFSAR Subsection 15.1.1, Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature, this event is bounded by the Excessive Increase in Secondary Steam Flow event described in UFSAR Subsection 15.1.3 described below. With the proposed changes, the power range high positive flux rate trip, and the overpower ΔT and overtemperature ΔT reactor trips, provide the required protection.

- As described in UFSAR Subsection 15.1.2, Feedwater System Malfunctions that Result in an Increase in Feedwater Flow, addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature. With the proposed changes, the power range high positive flux rate trip, and the overpower ΔT and overtemperature ΔT reactor trips, prevent a power increase that leads to a DNBR less than the safety analysis limit value.

- As described in UFSAR Subsection 15.1.3, Excessive Increase in Secondary Steam Flow, an excessive increase in secondary system steam flow (excessive load increase incident) results in a power mismatch between the reactor core power and the steam generator load demand. With the proposed changes, the power range high positive flux rate trip, and the overpower ΔT and overtemperature ΔT reactor trips, provide the required protection.

- As described in UFSAR Subsection 15.1.4, Inadvertent Opening of a Steam Generator Relief or Safety Valve, the most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. With the proposed changes, the power range high positive flux rate trip, and the overpower ΔT and overtemperature ΔT reactor trips, provide the necessary protection from an accidental depressurization of the main steam system.

- As described in UFSAR Subsection 15.1.5, Steam System Piping Failure, the steam release arising from a rupture of a main steam line results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. With the proposed changes, the power range high positive flux rate trip, and the overpower ΔT and overtemperature ΔT reactor trips, provide the necessary protection for a steam line rupture.

As described in UFSAR Section 15.4, Reactivity and Power Distribution Anomalies, a number of faults are postulated that result in reactivity and power distribution anomalies. Detailed analyses are evaluated for the events that have been identified as limiting cases to include the proposed changes to the power range high positive flux rate trip, and the overpower ΔT and overtemperature ΔT reactor trips, including the following:

- As described in UFSAR Subsection 15.4.1, Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical or Low-Power Startup Condition, an RCCA withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCAs which results in a power excursion. With the proposed changes, the power range high positive flux rate trip, and the source range high neutron flux, intermediate range high neutron flux, and power range high neutron flux (low setpoint) trips, continue to be assumed to be initiated and provide the required protection.

- As described in UFSAR Subsection 15.4.2, Uncontrolled RCCA Bank Withdrawal at Power, an uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. With the proposed changes, and to avert damage to the fuel cladding, the power range high positive flux rate trip, and the overpower ΔT and overtemperature ΔT reactor trips, continue to terminate any such transient before the DNBR falls below the design limit.

- As described in UFSAR Subsection 15.4.4, Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature, there will be no increase in core power, and no automatic or manual protective action is required. However, with the proposed changes,

the power range high positive flux rate trip would provide the required protection for this postulated event, backed up by the power range high neutron flux trip (high setting).

- As described in UFSAR Subsection 15.4.6, Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant, other than control rod withdrawal, the principal means of positive reactivity insertion to the core is the addition of unborated, primary-grade water from the demineralized water transfer and storage system into the reactor coolant system through the reactor makeup portion of the chemical and volume control system. Boron dilutions during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation are considered in this analysis. With the proposed changes, inadvertent boron dilution events are prevented during refueling and automatically terminated during cold shutdown, safe shutdown, and hot standby modes. In addition, inadvertent boron dilution events during startup or power operation, if not detected and terminated by the operators, result in an automatic overpower reactor trip by the power range high positive flux rate trip, backed up by the power range high neutron flux trip (high setting), with the proposed changes. Following reactor trip, automatic termination of the dilution occurs and post-trip return to criticality continues to be prevented.

- As described in UFSAR Subsection 15.4.8, Spectrum of Rod Cluster Control Assembly Ejection Accidents, this accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. With the proposed changes, the protection for this accident continues to be provided by the power range high positive flux rate trip, backed up by the power range high neutron flux trip (high setting).

<u>Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.2.1</u>

To address this issue, COL Appendix A Technical Specification 3.2.1 is proposed to be revised as follows:

- Required Action A.2 that requires the operator to “Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit” for the Condition of $F_Q^C(Z)$ not within limit as specified in the COLR is proposed to be deleted. In addition, Required Action B.2 that requires the operator to “Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit” for the Condition of $F_Q^W(Z)$ not within limits as specified in the COLR is proposed to be deleted. Required Action A.3 is not changed, and still requires the operator to “Reduce Overpower ΔT trip setpoint ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit” for the Condition of $F_Q^C(Z)$ not within limit as specified in the COLR. In addition, Required Action B.3 is not changed, and still requires the operator to “Reduce Overpower ΔT trip setpoint ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit” for the Condition of $F_Q^W(Z)$ not within limits as specified in the COLR. These are less restrictive changes that reflect the use of overpower ΔT trip for overpower protection instead of power range high neutron flux (high setpoint) trip for reactor protection during MSHIM operations. These proposed changes are acceptable, because for events that are not rapid, the overpower ΔT trip provides margin to the 118% RTP acceptance limit

as long as the setpoints are reduced to compensate for $F_Q^C(Z)$ or $F_Q^W(Z)$ exceeding the limits specified in the COLR. The power range high positive flux rate trip setpoints are not required to be changed, as the Required Actions A.1 and B.1 are not changed and still require the operator to reduce THERMAL POWER ≥ 1% for each 1% $F_Q^C(Z)$ or $F_Q^W(Z)$ exceeds limits. The reduction in THERMAL POWER alone is sufficient so that the power range high positive flux rate trip still prevents fuel design limits being exceeded during design basis events for those events where this reactor trip is applicable for overpower protection. The overlap between the responses of the power range high positive flux rate trip with reduced THERMAL POWER limits, and the responses of the overpower ΔT trip with reduced setpoint, in response to $F_Q^C(Z)$ or $F_Q^W(Z)$ exceeding the limits specified in the COLR, is acceptable to prevent exceeding fuel design limits during design basis events in MODE 1 with THERMAL POWER ≥ 25% RTP.

Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.2.2

To address this issue, COL Appendix A Technical Specification 3.2.2 is proposed to be revised as follows:

- Required Action A.1.2.2 that requires the operator to "Reduce Power Range Neutron Flux – High trip setpoints to ≤ 55%" for the Condition of $F^N_{\Delta H}$ not within limits as specified in the COLR is proposed to be revised to require the operator to "Reduce Overpower ΔT trip setpoints to ≤ 55%." This is a less restrictive change that reflects the use of overpower ΔT trip for overpower protection instead of power range high neutron flux (high setpoint) trip for reactor protection during MSHIM operations. For events that are not rapid, the overpower ΔT trip provides margin to the 118% RTP acceptance limit as long as the setpoints are reduced to compensate for $F^N_{\Delta H}$ exceeding the limits specified in the COLR. The overlap between the responses of the power range high positive flux rate trip, and the responses of the overpower ΔT trip with reduced setpoint, in response to $F^N_{\Delta H}$ exceeding the limits specified in the COLR, is acceptable to prevent exceeding fuel design limits during design basis events in MODE 1 with THERMAL POWER ≥ 25% RTP.

Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.7.1

To address this issue, COL Appendix A Technical Specification 3.7.1 is proposed to be revised as follows:

- Required Action A.2 that requires the operator to "Reduce the Power Range Neutron Flux – High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs." for the Condition of one or both steam generators with one or more MSSVs inoperable for opening is proposed to be revised to require the operator to "Reduce the Overpower ΔT reactor trip setpoints to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs." This is a less restrictive change that reflects the use of overpower ΔT trip for overpower protection instead of power range

high neutron flux (high setpoint) trip for reactor protection during MSHIM operations. For events that are not rapid, the overpower ΔT trip provides margin to the 118% RTP acceptance limit as long as the setpoints are reduced to compensate for the number of OPERABLE MSSVs available, and to prevent secondary system overpressurization for the limiting event occurring in MODE 1 that may result in secondary system overpressurization (i.e., an RCCA bank withdrawal at power event). The change from "setpoint" to "setpoints" is an administrative change for consistency with other references in the COL Appendix A Technical Specifications (e.g., COL Appendix A Technical Specifications 3.2.1 and 3.2.2), and recognizes that there are multiple divisions (four) of the overpower ΔT reactor trip function, each with an independent setpoint that must be reduced when required.

Description of any Changes to Current Licensing Basis Documents

COL Appendix A Technical Specifications Changes:

The following changes to the COL Appendix A Technical Specifications are proposed:

1. Technical Specification 3.2.1 REQUIRED ACTIONS A.2 and B.2 are deleted, and subsequent REQUIRED ACTIONS renumbered where applicable.

2. Technical Specification 3.2.2 REQUIRED ACTION A.1.2.2 is revised to replace "Power Range Neutron Flux – High" with "Overpower ΔT."

3. Technical Specification 3.7.1 REQUIRED ACTION A.2 is revised to replace "Power Range Neutron Flux – High reactor trip setpoint" with "Overpower ΔT reactor trip setpoints."

UFSAR Changes:

The following licensing basis changes to the UFSAR that are directly involved with the proposed changes to the COL Appendix A Technical Specifications or to the proposed safety analyses changes to credit the overpower ΔT trip, and for rapid power increases the power range high positive flux rate trip, instead of the power range high neutron flux (high setpoint) trip, are proposed:

1. UFSAR Subsection 7.2.1.1.2, Nuclear Overpower Trips, for Power Range High Neutron Flux Trip (High Setpoint), is revised to change the description for the power range high neutron flux (high setpoint) reactor trip, adding the statement that it provides a backup to the overtemperature ΔT, overpower ΔT, and power range high positive flux rate trips as it no longer provides primary protection in the safety analyses. However, the trip is not deleted from the design, and remains as always being active and required operable in the COL Appendix A Technical Specifications.

2. UFSAR Subsection 7.2.1.1.2, Nuclear Overpower Trips, for Power Range High Positive Flux Rate Reactor Trip, is revised to change the description for the power range high positive flux rate reactor trip as protecting the reactor against a rapid increase in core power generation during normal operation and as always being active, deleting the statement that this reactor trip provides protection against ejection accidents of low

worth rods from midpower as it provides protection for additional events at all power levels.

3. UFSAR Subsection 7.2.1.1.3, Core Heat Removal Trips, for Overpower ΔT Trip, is revised to delete the function of the Overpower ΔT reactor trip as a backup to the power range high neutron flux reactor trip.

4. UFSAR Subsection 15.0.6, Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses, is revised to add the statement "As described in Section 7.2 and in Reference 16 [Burnett, Toby, "Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips," APP-GW-GLR-137, Revision 1, February 2011], the overpower ΔT trip protects the core from exceeding the design overpower limit, and the overtemperature ΔT trip protects the core from exceeding the DNB design limit. As shown on the figure, the overtemperature ΔT setpoint plus the error allowances tracks the core DNB design limits, except that the setpoint includes an upper limit on allowable inlet temperature."

5. UFSAR Subsection 15.0.7, Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux, is revised as follows:

   a. Title is revised to delete ", Power Range Neutron Flux."

   b. First paragraph regarding discussion on instrumentation uncertainties and calorimetric uncertainties used in establishing the power range neutron flux setpoint is deleted.

   c. Second paragraph, second sentence, discussion on determining calorimetric uncertainty on a daily basis is revised to describe the methodology as comparing the secondary plant measurements with the ΔT power signal (Reference 16 [Burnett, Toby, "Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips," APP-GW-GLR-137, Revision 1, February 2011]) and with the total ion chamber current (sum of the top and bottom currents), and adjusting those signals if necessary for acceptable conformance with the calorimetric power measurement.

6. UFSAR Subsection 15.0.11.2, LOFTRAN Computer Code, first paragraph, sixth sentence, is revised to add "power range high positive flux rate" to the list of protection and safety monitoring system reactor trips included in the LOFTRAN computer code used for studies of transient response to specified perturbations in process parameters.

7. UFSAR Subsection 15.0.16, References, is revised to add Reference 16 as Burnett, Toby, "Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips," APP-GW-GLR-137, Revision 1, February 2011.

8. UFSAR Table 15.0-4a (Sheet 1), Protection and Safety Monitoring System Setpoints and Time Delay Assumed in Accident Analyses, is revised to replace Function "Reactor trip on power range high neutron flux, high setting" with "Reactor trip on power range

high positive flux rate," and to replace Limiting Setpoint Assumed in Analyses with "15% with 60-second time constant."

9. UFSAR Table 15.0-5, Determination of Maximum Power Range Neutron Flux Channel Trip Setpoint, Based on Nominal Setpoint and Inherent Typical Instrumentation Uncertainties, is deleted.

10. UFSAR Table 15.0-6 (Sheets 1, 3, and 4), Plant Systems and Equipment Available for Transient and Accident Conditions, are revised as follows:

    a. Reactor Trip Functions for Section 15.1, Feedwater system malfunctions that result in an increase in feedwater flow, are revised to include "Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, manual."

    b. Reactor Trip Functions for Section 15.1, Excessive increase in secondary steam flow, are revised to include "Power range high positive flux rate and high neutron flux."

    c. Reactor Trip Functions for Section 15.1, Inadvertent opening of a steam generator safety valve, are revised to include "Power range high positive flux rate and high neutron flux."

    d. Reactor Trip Functions for Section 15.1, Steam system piping failure, are revised to include "Power range high positive flux rate and high neutron flux."

    e. Reactor Trip Functions for Section 15.4, Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition, are revised to include "power range high positive flux rate."

    f. Reactor Trip Functions for Section 15.4, Uncontrolled RCCA bank withdrawal at power, are revised to include "power range high positive flux rate."

    g. Reactor Trip Functions for Section 15.4, Startup of an inactive reactor coolant pump at an incorrect temperature, are revised to include "Power range high neutron flux, power range high positive flux rate."

    h. Reactor Trip Functions for Section 15.4, Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant, are revised to include "Source range high neutron flux, power range high neutron flux, power range high positive flux rate."

    i. Reactor Trip Functions for Section 15.4, Spectrum of RCCA ejection accidents, are revised to include "Power range high neutron flux, power range high positive flux rate."

11. UFSAR Subsection 15.1.2.2.2, Results, for Feedwater System Malfunctions that Result in an Increase in Feedwater Flow, is revised as follows:

    a. The first paragraph, last sentence, is revised to add ", or by the power range high positive flux rate trip" to the reactor trips if the incident occurs with the unit just critical at no-load.

    b. The fifth paragraph, third sentence, is revised to change "high neutron flux trip setpoint" to "overpower ΔT trip setpoint."

12. UFSAR Subsection 15.1.3.1, Identification of Causes and Accident Description, for Excessive Increase in Secondary Steam Flow, is revised to add "Power range high positive flux rate" to the applicable PMS signals that provide protection against an excessive load increase accident.

13. UFSAR Subsection 15.1.4.1, Identification of Causes and Accident Description, for Inadvertent Opening of a Steam Generator Relief or Safety Valve, is revised to add "power range high positive flux rate" to the applicable PMS signals that provide protection from an accidental depressurization of the main steam system.

14. UFSAR Subsection 15.4.2.1, Identification of Causes and Accident Description, for Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power, is revised as follows:

    a. The first paragraph, third sentence, is revised to state: "Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in a violation of the DNB design basis or excessive linear power."

    b. The first paragraph, last sentence, is revised to state: "Therefore, to avert damage to the fuel cladding, the protection and safety monitoring system (PMS) is designed to terminate any such transient before the DNBR falls below the design limit (see Section 4.4) or the overpower limit is exceeded."

    c. The third paragraph, first bullet, first sentence, is revised to state: "Power range neutron flux instrumentation actuates a reactor trip if two out of four divisions exceed setpoint."

    d. The third paragraph, first bullet, is revised to reverse the two reactor trip functions and revised to state: "1. Reactor trip on power range high positive flux rate" and "2. Reactor trip on power range high neutron flux (high setpoint)."

    e. The third paragraph, first bullet, last paragraph, is revised to state: "The first trip protects the core when a sudden abnormal increase in power is detected in the power range neutron flux channel in two out of four PMS divisions. It provides protection against rapid reactivity insertion accidents and is always active. The second trip serves as a backup to the overpower ΔT reactor trip and is always active."

f. The third paragraph, second bullet, second sentence, is revised to state: "This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressurizer pressure to protect against violating the DNB design basis."

g. The seventh paragraph, first bullet, is revised to state: "Power range high positive flux rate (fixed setpoint)."

15. UFSAR Subsection 15.4.2.2.1, Method of Analysis, for Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power, is revised as follows:

    a. The second paragraph, third bullet, is revised to state: "The power range high positive flux rate trip is assumed to be actuated when the power range neutron flux changes at a rate higher than 15% per second with a 60 second rate-lag time constant. The overpower ΔT and overtemperature ΔT trips include adverse instrumentation and setpoint uncertainties. The delays for trip actuation assumed are given in Table 15.0-4a."

    b. The third paragraph is deleted and replaced with the following: "If RCCA movement causes an adverse effect on the axial core power distribution, then the overtemperature ΔT trip setpoint is decreased as necessary to maintain margin to the DNBR design limit, and the overpower ΔT trip setpoint is decreased as necessary to maintain margin to the overpower limit."

16. UFSAR Subsection 15.4.2.2.2, Results, for Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power, is revised as follows:

    a. A new first paragraph is added to state: "Three reactor trip functions were credited in the analyses to provide protection over the entire range of reactivity insertion rates. These are the power range high positive flux rate, overtemperature ΔT, and overpower ΔT trips."

    b. The existing first paragraph becomes the new second paragraph, and the second sentence is revised to state: "Reactor trip on power range high positive flux rate occurs shortly after the start of the transient."

    c. A new third paragraph is added to state: "The transient response for a representative intermediate (34 pcm/s) RCCA withdrawal from full power is shown in Figures 15.4.2-7 through 15.4.2-12. Reactor trip on overpower ΔT occurs preventing the peak heat flux from exceeding 118%. The DNB design basis described in Section 4.4 is met."

    d. The existing second paragraph becomes the new fourth paragraph, and the references to "Figures 15.4.2-7 through 15.4.2-12" are revised to "Figures 15.4.2-13 through 15.4.2-18."

    e. The existing third paragraph becomes the new fifth paragraph, and the reference in the first sentence to "Figure 15.4.2-13" is revised to "Figure 15.4.2-19," a comma is deleted in the second sentence, the fourth sentence is revised to state: "These are

the power range high positive flux rate, overpower ΔT, and overtemperature ΔT trip functions." and the last sentence is deleted.

f. The existing fourth paragraph becomes the new sixth paragraph, and the references in the first sentence to "Figures 15.4.2-14 and 15.4.2-15" are revised to "Figure 15.4.2-20 and 15.4.2-21," and the second, third, and fourth sentences are revised to state: "Minimum DNBR occurs immediately after rod motion. The results are similar to the 100-percent power case, except the transient is terminated by either the power range high positive flux rate or overtemperature ΔT reactor trip. The minimum DNBR is greater than the design limit value described in Section 4.4."

g. The existing sixth paragraph becomes the new eighth paragraph, and the first sentence is revised to state: "Referring to Figure 15.4.2-19, it is noted that for transients initiated from full power, three reactor trip functions provide the DNB and overpower protection over the range of reactivity insertion rates analyzed. The overtemperature ΔT trip provides DNB protection except for rapid power excursions. The overpower ΔT trip provides protection for the slow to moderate power excursions. The power range high positive flux rate trip prevents both overpower and low DNBR for rapid power excursions."

h. The subparagraphs A, B, C, and D under the new eighth paragraph are deleted.

i. The existing seventh and eighth paragraphs are deleted.

j. The tenth paragraph, second sentence, is revised to state: "For rapid power excursions, the overpower transient is fast with respect to the fuel rod thermal time constant and the core heat flux lags behind the neutron flux response."

k. The eleventh paragraph, first and second sentences, are revised to state: "For slow to moderate power excursions, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by either the overpower ΔT or overtemperature ΔT reactor trip before the DNB design basis is violated."

l. The twelfth paragraph, first sentence, is revised to state: "The reactor is tripped during the RCCA bank withdrawal at-power transient such that the ability of the primary coolant to remove heat from the fuel rods is not reduced."

17. UFSAR Subsection 15.4.2.3, Conclusions, first sentence, is revised to state: "The overpower ΔT, overtemperature ΔT, and power range high positive flux rate trip functions provide adequate protection over the entire range of possible reactivity insertion rates."

18. UFSAR Table 15.4-1, Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies, is revised as follows:

    a. For the Uncontrolled RCCA bank withdrawal at power accident, Item 1 Case A, the second event is revised to state: "Power range high positive flux rate trip point

reached" and the time is revised from "6.2" to "5.2" seconds. In addition, the next two events are revised from "7.1" to "6.1" seconds and "7.4" to "6.4" seconds, respectively.

b. For the Uncontrolled RCCA bank withdrawal at power accident, Item 2 Case B is renumbered to Item 3 Case C.

c. For the Uncontrolled RCCA bank withdrawal at power accident, a new Item 2 Case B is added including the following events and times:

- New event "Initiation of uncontrolled RCCA withdrawal at an intermediate reactivity insertion rate (34 pcm/s)" with a time of "0.0" seconds.
- New event "Overpower ΔT setpoint reached" with a time of "18.0" seconds.
- New event "Rods begin to fall into core" with a time of "19.9" seconds.
- New event "Minimum DNBR occurs" with a time of "20.1" seconds.

19. UFSAR Figures 15.4.2-1 through 15.4.2-6 for Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s) (corresponding to the existing Uncontrolled RCCA bank withdrawal at power accident, Item 1 Case A, in UFSAR Table 15.4-1) are updated to reflect the latest analysis results.

20. New UFSAR Figures 15.4.2-7 through 15.4.2-12 for Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s) (corresponding to the new Uncontrolled RCCA bank withdrawal at power accident, Item 2 Case B, in UFSAR Table 15.4-1) are added to reflect the latest analysis results.

21. UFSAR Figures 15.4.2-7 through 15.4.2-12 for Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s) (corresponding to the renumbered Uncontrolled RCCA bank withdrawal at power accident, Item 3 Case C, in UFSAR Table 15.4-1) are renumbered as UFSAR Figures 15.4.2-13 through 15.4.2-18 and updated to reflect the latest analysis results.

22. UFSAR Figure 15.4.2-13 for Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 100-percent Power is renumbered as UFSAR Figure 15.4.2-19 and updated to reflect the latest analysis results.

23. UFSAR Figure 15.4.2-14 for Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 60-percent Power is renumbered as UFSAR Figure 15.4.2-20 and updated to reflect the latest analysis results.

24. UFSAR Figure 15.4.2-15 for Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 10-percent Power is renumbered as UFSAR Figure 15.4.2-21 and updated to reflect the latest analysis results.

2.3 Recovery of Overpower ΔT and Overtemperature ΔT Reactor Trip System Setpoint Margins

The AP1000 plant reactor protection system setpoint uncertainty analysis, using the original uncertainty inputs for the overpower ΔT and overtemperature ΔT trip setpoints, results in negative setpoint margin for these functions. For the overpower ΔT trip function, using the total allowance of 6.0% RTP and an uncertainty of 13.3% RTP results in a negative setpoint margin of -7.3% RTP. For the overtemperature ΔT trip function, using the total allowance of 5.6% RTP and an uncertainty of 6.5% RTP results in a negative setpoint margin of -0.9% RTP. The acceptance criterion for each function is setpoint margin ≥ 0.0% RTP. Therefore, setpoint margin must be recovered to obtain acceptable results for the overpower ΔT and overtemperature ΔT trip functions.

COL Appendix A Technical Specification 3.3.1 is proposed to be revised to address this issue as follows:

- SR 3.3.1.4 requires comparison of the incore detector measurements to nuclear instrument channel AFD every 31 effective full power days (EFPD). Note 1 modifying SR 3.3.1.4 requires adjustment of the nuclear instrument channel in PMS if absolute difference is ≥ 3% AFD. However, this value is proposed to be reduced to ≥ 1.5% AFD to provide for partial recovery of setpoint margin to obtain acceptable results for the overpower ΔT and overtemperature ΔT trip functions.

To provide for partial recovery of setpoint margin to obtain acceptable results for the overpower ΔT and overtemperature ΔT trip functions, the absolute difference between the incore detector measurements to nuclear instrument channel AFD obtained during performance of SR 3.3.1.4 at which adjustments are required to be made is proposed to be reduced from ≥ 3% AFD to ≥ 1.5% AFD. Additional changes to the penalty slopes for the overpower ΔT and overtemperature ΔT trip functions are proposed to provide for additional recovery of setpoint margin. In addition, an insulation resistance degradation error is another design input for the overpower ΔT nominal trip setpoint. The insulation resistance degradation error calculation includes a resistance temperature detector (RTD) cable splice insulation resistance value of 1.0E6 ohms (based on the procurement specification acceptance criteria), and results in an insulation resistance degradation error of 1.7°F. This insulation resistance error is a significant input to the overpower ΔT trip setpoint for a steam line break. Therefore, to complete recovery of setpoint margin, the overpower limit for events with a harsh environment is increased from 118% RTP to 119% RTP, and the overpower ΔT nominal trip setpoint is reduced from 109% RTP to 108.5% RTP.

This activity affects the PMS, including the reactor protection system. The reactor protection system is designed to actuate a reactor trip whenever necessary to prevent exceeding the fuel design limits. The core design, together with the process and decay heat removal systems, provide this capability under expected conditions of normal operation, with appropriate margins for uncertainties and anticipated transient situations.

As described in COL Appendix C and corresponding plant-specific Tier 1 Subsection 2.5.2, Protection and Safety Monitoring System, the PMS initiates an automatic reactor trip, as identified in COL Appendix C (and plant-specific Tier 1) Table 2.5.2-2, PMS Automatic

Reactor Trips, when plant process signals reach specified limits. The PMS has the equipment identified in COL Appendix C (and plant-specific Tier 1) Table 2.5.2-1, PMS Equipment Name and Classification. The PMS has four divisions of reactor trip. Setpoints are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation. The relevant PMS automatic reactor trips include the overpower ΔT trip, and the overtemperature ΔT trip.

Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.3.1

To address this issue, COL Appendix A Technical Specification 3.3.1 is proposed to be revised as follows:

- Note 1 modifying SR 3.3.1.4 is proposed to be revised to change the criteria for when the nuclear instrument channel is to be adjusted from "if absolute difference is ≥ 3% AFD" to "if absolute difference is ≥ 1.5% AFD" as a more restrictive change. This proposed change is necessary to provide for partial recovery of setpoint margin to obtain acceptable results for the overpower ΔT and overtemperature ΔT trip functions.

There are notable differences between the current operating fleet and the AP1000 plant for the overpower ΔT and overtemperature ΔT trip functions, including the following:

- The AP1000 plant uses an improved digital design philosophy for the overpower ΔT and overtemperature ΔT trip functions. The operating fleet uses an older analog design which precludes a direct comparison of the measured core power to the core thermal limits. In the analog design, core limits must be converted into an equivalent power level which is compared to ΔT and reactor coolant system (RCS) average coolant temperature (T_{avg}) as derived by RCS hot leg temperature (T_{hot}) and RCS cold leg temperature (T_{cold}) RTD measurements. The AP1000 plant digital design provides for a simpler and more direct conversion of the core thermal limits to the protection setpoints and has a different dynamic compensation scheme, and improves available operating margin. This margin has been allocated to allow for operating at a higher rated thermal power than would be allowed with an analog protection system. However, these design differences result in changes to the design input parameters for the setpoint uncertainty analysis, the safety analysis limits, and the margin to trip analysis. Many of the typical instrument error allowances are removed and are replaced with the proposed changes to Note 1 modifying SR 3.3.1.3 to require comparison of the calorimetric heat balance to the ΔT power calculation ($q_{\Delta T}$) output if absolute difference between $q_{\Delta T}$ and the calorimetric measurement is > 3% RTP.

- The current operating fleet is designed and typically operated at base load with all control rods out. Little to no load follow consideration is designed into the fuel cycle or plant control/protection system setpoints. For the AP1000 plant design, MSHIM load follow operation is part of the plant design basis, and both base load and load follow operations rely on significant control rod insertion. This allowed for simplification of the chemical and volume control system and improved load follow characteristics. Control rod insertion is expected to influence indicated hot leg temperatures due to increased

variations in hot leg streaming. These design conditions are accounted for in the UFSAR Chapter 15 safety analysis.

- In both the operating fleet and AP1000 plant design, the overpower ΔT trip function is credited for protection of the UFSAR Chapter 15 safety analysis for intermediate-sized steam line breaks. As a result of a steam line break event, instruments can be affected by elevated temperatures inside containment. Insulation resistance degradation of the in-containment cables, connectors, splices, and penetration must be considered. The AP1000 plant design input for insulation resistance is larger than what is typically used in the US operating fleet. Typically US operating fleet allowances are approximately 0.5°F on indicated ΔT. The AP1000 current plant design input is 1.7°F.

- For the AP1000 plant design, the overpower ΔT nominal trip setpoint requires an f(ΔI) penalty to support the core thermal power limits analysis. This penalty is generally not required for the current operating plants for the overpower ΔT trip function, but is necessary for AP1000 plants because of the higher plant power density. The f(ΔI) penalty input to the trip function is based on the Nuclear Instrumentation System (NIS) excore power range detectors. As a result, the uncertainty analysis must account for the COL Appendix A Technical Specification SR 3.3.1.4 allowance of ±3% AFD for incore/excore AFD.

Currently, COL Appendix A Technical Specification SR 3.3.1.4 requires that AFD agree within ±3%. This value has historically been associated with operating plants that calibrate the excore detectors to the core average ΔΦ, as measured by the incore detectors. In the AP1000 plant design, the excore detectors are proposed to be calibrated to the WPA ΔΦ as previously described, because the core average ΔΦ measured by the excore detectors is subject to increased uncertainty, due to the effect of control rod motion associated with the MSHIM operating strategy. Since the excore detectors are primarily exposed to neutrons leaking out of the core from the nearest peripheral fuel assemblies, the excore indicated power and AFD closely track the weighted peripheral assembly power and AFD values, without requiring frequent adjustments in the calibration. This is known to be true, because the excore detectors respond predominantly to the neutrons sourced from these specific assemblies. Therefore, changes in power and power shape in these assemblies are accurately reflected in the excore response. This coupling is only affected by changes in the core makeup (e.g., flux suppression rods) or in the makeup of the detector or cavity geometry (e.g., moderator sleeves). Those changes do not occur during a cycle of operation and are reflected in the plant modification control process.

WPA ΔΦ is different from core average ΔΦ mainly due to the lower relative power generated by the peripheral fuel assemblies that are seen by the excore detectors. In the low leakage core designs expected to be used at AP1000 plants, these assemblies generate less than 50% of the core average assembly power. Low leakage core designs are more economical, and increase the difference between peripheral and core average AFD. This fact makes WPA ΔΦ values less than 50% of core average ΔΦ values for identical core states.

Until now, it has been conservatively interpreted that the SR 3.3.1.4 ±3% AFD requirement still applies to the calibrated WPA ΔΦ. In fact, due to the known stability of the weighted

peripheral calibration, the ±3% AFD acceptance criterion is larger than it needs to be. The proposal is for AP1000 plants to change the SR 3.3.1.4 limit from ±3% AFD to ±1.5% AFD, and specify in the Technical Specifications Bases that this is based on calibrating the excore detectors to WPA ΔΦ.

SR 3.3.1.4 is required to be performed every 31 EFPD. Since AP1000 plants have fixed incore detectors (as opposed to moveable), the surveillance could be performed more often, based on the continuously measured incore power distribution. However, this frequency is sufficient to support operability of the overpower ΔT and overtemperature ΔT trip functions. In addition, the probability of meeting the ±1.5% AFD criterion is increased by this proposed change for the following reasons:

1. WPA ΔΦ is proportional to core average ΔΦ, but is a much smaller number because of the lower relative power being generated in the peripheral fuel assemblies. In Cycle 1, the average WPA ΔΦ is only approximately 37% to approximately 46% of core average ΔΦ, with the higher percentages always tending to occur toward the end of the fuel cycle, as burnup effects lead to more relative power generation in peripheral fuel assemblies. In the low leakage reload cycles studied to date for AP1000 plants, the average peripheral fuel assembly relative power seen by the excore detectors is predicted to be slightly lower than the Cycle 1 core, making Cycle 1 the limiting core in terms of assessing the effect of the WPA ΔΦ versus core average ΔΦ relationship on SR 3.3.1.4 compliance.

   Thus for example, if the core average ΔΦ is -10%, one would expect the WPA ΔΦ to be approximately -4.6% at the end of Cycle 1. Before the proposed change to calibrate the power range detector AFD inputs to the WPA ΔΦ, one hypothetical example would be where the incore detector core average ΔΦ is -10% and the excore detector core average ΔΦ is reading -7% due to a 30% calibration error (i.e., the excore detector is only indicating 70% of the incore core average ΔΦ).

   This hypothetical example corresponds to a 3% difference in relation to meeting SR 3.3.1.4 acceptance, and would result in just meeting the SR 3.3.1.4 acceptance criterion. After the proposed change to calibrate the power range detector AFD inputs to the WPA ΔΦ, the same situation would result in an incore WPA ΔΦ of about -4.6% (worst case Cycle 1 burnup) and the excore detectors (reading 70% of that) would read -3.22% for the same level of decalibration. This corresponds to a 1.38% difference in relation to SR 3.3.1.4. Thus, a situation that would have just met the SR 3.3.1.4 acceptance criterion prior to the proposed change to calibrate the power range detector AFD inputs to the WPA ΔΦ would now satisfy the SR 3.3.1.4 acceptance criterion, even if the acceptance criterion is reduced from 3% AFD to 1.5% AFD. In this example, the margin to the SR 3.3.1.4 acceptance criterion has increased, just due to the average relationship between core average ΔΦ and WPA ΔΦ.

2. The excore detectors naturally only detect the power distribution on a few peripheral fuel assemblies. This means that any axial or radial shift in the peripheral assembly power distribution caused by the movement of control rods will be seen in both the incore power distribution WPA ΔΦ and the excore calibrated WPA ΔΦ, by approximately the same amounts. The uncertainty in the excore detector WPA ΔΦ

reading due to rod shadowing becomes zero if calibrated to incore WPA ΔΦ instead of core average ΔΦ. Prior to the proposed change to calibrate the power range detector AFD inputs to the WPA ΔΦ, the movement of control rods could affect the incore core average ΔΦ and the excore readings (which were scaled up readings of WPA ΔΦ) in different manners. Calculations indicated that these rod shadowing induced variations could be as high as ±10% in the indicated excore core average ΔΦ (which would have directly affected the setpoint uncertainty calculations, and resulted in frequent challenges to the SR 3.3.1.4 ±3% AFD acceptance criterion). By choosing to calibrate the excore detectors to WPA ΔΦ, the rod shadowing uncertainty is zeroed out for excore measurements, and incorporates its effects directly into the 3D FAC power distribution analysis, where the trip setpoint f(ΔI) penalty function generator is determined in a manner that conservatively bounds all the rod shadowing effects.

An evaluation has been completed that quantifies the standard deviation of the excore detector calibration made to the weighted peripheral AO between 0.444% and 0.612%, which is expected to bound the weighted peripheral AFD standard deviation. This is an indicator that the SR 3.3.1.4 results are expected to easily meet the revised acceptance criterion based on the weighted peripheral calibration.

Technical Evaluation of Proposed Changes to f(ΔI) Penalty Slopes for Overpower ΔT and Overtemperature ΔT Trip Functions

In the revised 3D FAC power distribution analysis for AP1000 plant Cycle 1 cores, the f(ΔI) penalty function generator slopes and deadbands have been revised to generate more operating space for normal operation and Condition II accidents (or faults of moderate frequency) and to reduce the uncertainty components associated with the slope of the f(ΔI) penalty function generator. The revised overpower ΔT and overtemperature ΔT trip functions make spurious actuations of low margin alarms and turbine runbacks less likely to occur during normal operation. The new penalty functions are made possible by reducing the excess conservatism in the transient xenon distributions considered in the 3D FAC power distribution analysis, and by separately analyzing both control rod insertion sequences. In addition, the new penalty functions incorporate only the applicable Condition II accident setpoint uncertainties, and do not include excess uncertainty to account for containment harsh environment, as might be expected during a more severe accident (e.g., steam line break) inside containment.

Technical Evaluation of Proposed Changes to Overpower Limit for Events with a Harsh Environment and Overpower ΔT Nominal Trip Setpoint

Three unique cases must be considered for the overpower ΔT nominal trip setpoint uncertainty calculation. Note: The cases described below are based on the current overpower ΔT nominal trip setpoint of 109% RTP and do not incorporate the proposed changes described herein.

1. Case #1 considers normal and harsh environment uncertainties and no consideration for f(ΔI) penalty function generator uncertainties. This case demonstrates that the safety analysis value for the overpower ΔT setpoint of 115% RTP is not satisfied for the defined uncertainties.

2. Case #2 considers normal uncertainties (no harsh environment uncertainties) and f(ΔI) penalty function generator uncertainties. This case demonstrates that the safety analysis value for the overpower ΔT setpoint of 115% RTP is not satisfied with an axial flux difference.

3. Case #3 considers normal and harsh environment uncertainties due to steam line break and f(ΔI) penalty function generator uncertainties. This case demonstrates that the overpower limit of 118% RTP for events with a harsh environment is not satisfied for the defined uncertainties.

With a large insulation resistance degradation error (1.7°F) the most limiting case is Case #3. To obtain positive margin, the overpower limit for events with a harsh environment is increased from 118% RTP to 119% RTP to address Case #3, and the overpower ΔT nominal trip setpoint is reduced from 109% RTP to 108.5% RTP to address these three cases.

The full power steam line break analysis peak power is increased from 118% RTP to 119% RTP to account for the additional overpower ΔT trip setpoint uncertainty associated with the harsh containment environment. This represents a revision to an internal Westinghouse analysis which currently contains approximately 13% margin to the applicable design basis peak kW/ft limit. The revised results meet the same kW/ft limit. The value of the peak analyzed power level and the calculated peak kW/ft are not reported as specific values in the licensing basis.

Changing the overpower ΔT nominal trip setpoint from 109% RTP to 108.5% RTP, the following design basis operational transients have been analyzed for the AP1000 plant:

- 10% Step Increase and Decrease in Turbine Load
- Turbine Loading and Unloading at 5% per minute
- Load Rejections up to a Full Load Rejection
- Turbine Trip
- Main Feedwater Pump Trip
- Load Follow
- Frequency Control
- Control Rod Sequence Exchange

These transients represent the plant design basis and are generally more severe than those experienced during normal operation. For example, turbine loading is limited to 5% per minute and is generally performed at a much slower rate. The transients resulting in a turbine load decrease (load rejections, turbine trip, main feed water pump trip, etc.), decrease the ΔT via an initial cold leg temperature increase and core power reduction. Therefore these transients do not challenge the overpower ΔT trip setpoint and margin

generally increases from the full power steady-state value. The most limiting of these transients with respect to margin to the overpower ΔT trip setpoint are those that result in a turbine load increase to full power (e.g., loading at 5% per minute and 10% step load increase). The results of the analyses show that the minimum margin to the revised overpower ΔT nominal trip setpoint of 108.5% RTP is greater than approximately 3.6% RTP, which is acceptable.

The control rod sequence exchange transient can also be limiting to the overpower ΔT trip setpoint, depending on the initial control bank positions. Since the automatic axial offset control system is disabled during the exchange, the motion of the exchanging banks can shift the AFD significantly enough to reach the overpower ΔT trip setpoint due to the f(ΔI) penalty function generator if the control rod exchange is initiated at full power with deeply inserted banks. However, the initial control bank positions are procedurally controlled such that a control rod sequence exchange will not be performed from adverse conditions. Furthermore, the revised f(ΔI) penalty function generator discussed above provides a benefit that is not captured in the current analysis.

There are various control functions that act to reduce the likelihood of a plant trip on overpower ΔT during normal operation when low margin is detected. These include a turbine loading suspension, a block on automatic and manual rod withdrawal, and a turbine runback. These functions are actuated on low (3% RTP) and low-low (1% RTP) margin to the overpower ΔT trip function and also have corresponding alarms. The low and low-low margin setpoints change proportionally with the trip setpoints. While these functions are intended to mitigate low margin to trip conditions, they can also result in a nuisance to operators if spuriously actuated.

The first function to actuate is the turbine loading suspension and the corresponding alarm on low margin to the overpower ΔT trip setpoint when the margin to the overpower ΔT nominal trip setpoint decreases to 3% RTP. As noted above, the minimum margin for the limiting design basis operational transients was approximately 3.6% RTP, which remains greater than this function setting. Furthermore, the impact of a hypothetical fluctuation in T_{hot} was evaluated via an informal analysis. The results of this analysis show that the overpower ΔT nominal trip setpoint margin remains greater than the 3% RTP low margin setting. Therefore, the reduction in the overpower ΔT nominal trip setpoint to 108.5% RTP will not result in nuisance alarms and will not result in spurious plant trips during normal operation.

It should be noted that an insulation resistance degradation error is only needed for functions that are credited for harsh environment conditions. The overpower ΔT trip function is credited for an intermediate steam line break and must account for a harsh environment. The overtemperature ΔT trip function is not credited for harsh environment conditions and does not need to include an insulation resistance degradation error. Therefore, to obtain positive margin for the overpower ΔT trip function, a change to the incore/excore AFD limit from 3% AFD to 1.5% AFD and a change to the f(ΔI) penalty function generator is still required.

Conclusion

To recover setpoint margin for the overpower ΔT and overtemperature ΔT trip functions, a solution is developed resulting in minimal impact to the AP1000 plant design and plant operations. The solution consists of three aspects:

1. Change the Technical Specifications incore/excore AFD limit from 3% AFD to 1.5% AFD.

2. Change the f(ΔI) penalty function generator for the overpower ΔT and overtemperature ΔT trip functions.

3. Change the overpower limit for events with a harsh environment from 118% RTP to 119% RTP (for Case #3) and reduce the overpower ΔT nominal trip setpoint from 109% RTP to 108.5% RTP.

The effect of this change to the operation of the plant was evaluated and concluded that the changes pose a minor impact.

Calibrating the excore detectors to the WPA ΔΦ is expected to be much more stable than previously calibrating the excore detectors to the core average ΔΦ due to MSHIM operation. The current SR 3.3.1.4 acceptance criterion of ±3% AFD was based on calibrating the excore detectors to the core average ΔΦ. With the increased stability of using WPA ΔΦ, the SR 3.3.1.4 acceptance criterion is overly conservative and can be reduced to ±1.5% AFD. Supporting justification for this change has been provided above, and an evaluation has been completed that quantifies the standard deviation of the excore detector calibration made to the weighted peripheral AO between 0.444% and 0.612%, which is expected to bound the weighted peripheral AFD standard deviation. This is an indicator that the SR 3.3.1.4 results are expected to easily meet the revised acceptance criterion based on the weighted peripheral calibration.

Revising the f(ΔI) penalty function generator for the overpower ΔT and overtemperature ΔT trip functions generates additional operating space for normal operation and Condition II transients. This is possible by reducing conservatisms in the 3D FAC power distribution analysis and analyzing both control rod insertion sequences separately.

Increasing the overpower limit for events with a harsh environment to 119% RTP, and reducing the overpower ΔT nominal trip setpoint to 108.5% RTP, are also acceptable. The change in the overpower limit for events with a harsh environment from 118% RTP to 119% RTP represents a revision to an analysis which currently contains approximately 13% RTP margin, and the revised results are expected to easily meet the same limits. The value of the peak analyzed power level and the calculated peak kW/ft are not reported as specific values in the licensing basis. The nuclear steam supply system control system analysis was also evaluated, and it was determined that a reduction of the overpower ΔT nominal trip setpoint to 108.5% RTP still provides a minimum margin of approximately 3.6% RTP for the limiting design basis operational transient. This margin is still greater than the 3% RTP low margin setting that triggers automatic actions to reduce the likelihood of a plant trip on overpower ΔT during normal operation.

In summary, the proposed solution to recover margin for the overpower ΔT and overtemperature ΔT trip functions has been evaluated, and it is concluded that the overall impact to the AP1000 plant design and plant operations is minimal.

Based on an analysis of the actions proposed, it is estimated that the final setpoint margin for the overpower ΔT trip function will be approximately 0.1% RTP, and for the overtemperature ΔT trip function will be approximately 0.5% RTP. Although this results in a reduction in operating margin, an assessment of this impact is that these changes will not significantly impact plant operations.

The UFSAR Chapter 15 safety analyses are not impacted by these changes to the overpower limit for events with a harsh environment to 119% RTP, and reduction of the overpower ΔT nominal trip setpoint to 108.5% RTP, because the changes do not affect any of the inputs, methodology, assumptions, or acceptance criteria that are modeled in the safety analyses reported in the licensing basis. No changes to the UFSAR Chapter 15 safety analyses result from these changes.

Description of any Changes to Current Licensing Basis Documents

COL Appendix A Technical Specifications Changes:

The following changes to the COL Appendix A Technical Specifications are proposed:

1. Technical Specification 3.3.1 SURVEILLANCE REQUIREMENT SR 3.3.1.4 NOTE 1 is revised to replace "≥ 3% AFD" with "≥ 1.5% AFD" for the absolute AFD difference that requires the nuclear instrumentation channel to be adjusted when comparing the results of the incore detector measurements to nuclear instrument channel AFD.

UFSAR Changes:

The following licensing basis changes to the UFSAR are proposed:

None.

2.4 Changes to COL Appendix A Technical Specification 3.1.3, Moderator Temperature Coefficient (MTC)

This activity affects monitoring of the MTC of the AP1000 plant reactor core design. As described in UFSAR Subsection 4.3.1.2, for the initial fuel cycle, the fuel temperature coefficient will be negative, and the MTC of reactivity will be negative for power operating conditions, thereby providing negative reactivity feedback characteristics. The design basis meets General Design Criterion (GDC) 11.

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption (Doppler) effects associated with changing fuel temperature and the neutron spectrum and reactor composition change effects resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium results in a Doppler coefficient of reactivity that is negative. This coefficient provides the most rapid reactivity

compensation. The initial core is also designed to have an overall negative MTC of reactivity during power operation so that average coolant temperature changes or void content provides another, slower compensatory effect. For some core designs, if the compensation for excess reactivity is provided only by chemical shim, the MTC could become positive. Nominal power operation is permitted only in a range of overall negative MTC. The negative MTC can be achieved through the use of discrete burnable absorbers and/or integral fuel burnable absorbers and/or control rods by limiting the reactivity controlled by soluble boron.

Burnable absorber content (quantity and distribution) is not stated as a design basis. However, for some reloads, the use of burnable absorbers may be necessary for power distribution control and/or to achieve an acceptable MTC throughout core life. The required burnable absorber loading is that which is required to meet design criteria.

As described in UFSAR Subsection 4.3.2.1, the nuclear design includes the use of soluble neutron absorber (i.e., boron), which is varied to compensate for reactivity changes due to fuel burnup, fission product poisoning including xenon and samarium, burnable absorber depletion, and the cold-to-operating moderator temperature change. Throughout the operating range, the chemical and volume control system is designed to provide changes in reactor coolant system boron concentration to compensate for the reactivity effects of fuel depletion, peak xenon burnout and decay, and cold shutdown boration requirements.

The means for predicting and maintaining MTC within the required limits are described in COL Appendix A Technical Specification 3.1.3. MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the MTC is less than zero at 100% of RTP. The actual value of the MTC is dependent on core characteristics such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons (burnable absorbers) to yield an MTC within the range analyzed in the plant accident analysis. The EOL MTC is also limited by the requirements of the accident analysis. Fuel cycles designed to achieve high burnups that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOL limit. The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR Chapter 15 accident and transient analyses. The COL Appendix A Technical Specification 3.1.3 Surveillance Requirements (SRs) for measurement of the MTC at BOL and near EOL for each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reactor coolant system boron concentration changes associated with fuel burnup and burnable absorbers depletion.

2.4.1 Measurement of the Moderator Temperature Coefficient (MTC) at Beginning of Life (BOL)

COL Appendix A Technical Specification SR 3.1.3.1 requires measurement of the MTC once at BOL during each fuel cycle prior to entering MODE 1 in order to demonstrate compliance with the MTC upper limit as specified in the COLR. Meeting the limit prior to entering MODE 1 assures that the limit will also be met at higher power levels. Based on the use of MSHIM automatic operations for reactivity control, in conjunction with manual

operation of the chemical and volume control system for boration control, administrative limits based on a combination of control bank position and maximum RCS boron concentration are proposed for ensuring the MTC upper limit during each fuel cycle is met. The critical boron concentration may increase with burnup early in the cycle if the core design includes a large amount of boron absorber, which impacts the verification that the MTC upper limit is met not only at BOL but also throughout the fuel cycle. Burnable absorbers are designed to suppress the critical boron concentration early in a fuel cycle in order to control the MTC. As the fuel cycle progresses, the burnable absorbers are designed to deplete, so that the neutron absorption from the burnable absorbers is minimal at EOL in order to optimize the power generation cost of the fuel cycle.

The COL Appendix A Technical Specifications place constraints on the MTC based on the accident analysis assumptions for the moderator density coefficient. To demonstrate compliance with the MTC upper limit (i.e., most positive MTC), SR 3.1.3.1 requires measurement of the MTC once at BOL for comparison with the MTC upper limit as specified in the COLR.

Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.1.3

To address this issue, COL Appendix A Technical Specification 3.1.3 Required Action A.1 is proposed to be revised to allow the use of administrative limits based on a combination of control bank position and maximum RCS boron concentration for maintaining compliance with the MTC upper limit. Specifically, COL Appendix A Technical Specification Required Action A.1 is proposed to be revised to simply state “Restore MTC within limit” as a less restrictive change. The proposed administrative limits for maintaining compliance with the MTC upper limit consists of tables of maximum allowed RCS boron concentration versus control rod position, core power, and cycle burnup; and are determined on a cycle-specific basis by considering the margin predicted to the MTC upper limit.

Currently, the COL Appendix A Technical Specifications require measurements of MTC during each fuel cycle at BOL to verify the most positive MTC upper limit. At BOL, the measurement of the isothermal temperature coefficient is relatively simple to perform since it is done at hot zero power isothermal conditions and is not complicated by changes in the enthalpy rise or the presence of xenon. The use of the proposed administrative limits for ensuring the MTC upper limit is met through use of the combination of control bank position and maximum RCS boron concentration is acceptable as the new administrative limits provide acceptable margin predicted to the MTC upper limit during each fuel cycle. The SR 3.1.3.1 measurement results and predicted design value are used to determine if these proposed administrative limits on the maximum boron concentration versus control rod position, power level, and cycle burnup are required for compliance with the MTC upper limit. Plant safety criteria are not compromised by the use of these administrative limits.

Description of any Changes to Current Licensing Basis Documents

COL Appendix A Technical Specifications Changes:

The following changes to the COL Appendix A Technical Specifications are proposed:

1. Technical Specification 3.1.3 is revised as follows:

    a. REQUIRED ACTION A.1 is revised to replace "Establish administrative withdrawal limits for control banks to maintain" with "Restore" for the required action necessary to restore MTC within upper limit.

2.4.2 Measurement of the Moderator Temperature Coefficient (MTC) at End of Life (EOL)

COL Appendix A Technical Specification SR 3.1.3.2 requires the verification that MTC is maintained within the MTC lower limit as specified in the COLR for full power conditions near the EOL during each fuel cycle. However, SR 3.1.3.2 does not allow the use of an alternative NRC approved methodology described in WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement." This methodology allows replacement of the requirement to measure the MTC after reaching the equivalent of an equilibrium RTP (also described as Hot Full Power (HFP) in WCAP-13749-P-A) all rods out (ARO) boron concentration of 300 ppm near EOL, with a revised prediction of MTC at EOL using an algorithm from WCAP-13749-P-A.

The COL Appendix A Technical Specifications place constraints on the MTC based on the accident analysis assumptions for the moderator density coefficient. A positive moderator density coefficient corresponds to a negative MTC. To demonstrate compliance with the MTC lower limit (i.e., most negative MTC), SR 3.1.3.2 requires measurement of the MTC at any thermal power, but with the results extrapolated to the conditions of RTP (i.e., HFP) and all control banks withdrawn (i.e., ARO), in order to make a proper comparison with the MTC lower limit as specified in the COLR. Because the MTC value at HFP gradually becomes more negative with core depletion and boron concentration reduction during each fuel cycle, a boron concentration limit of 300 ppm maintains an MTC that is less negative than the MTC lower limit at EOL. The conditional exemption from performing the EOL MTC measurement is determined on a cycle-specific basis by considering the margin predicted to the MTC lower limit and the performance of other core parameters, such as BOL MTC measurements and the critical boron concentration as a function of fuel cycle length in accordance with the alternative NRC approved methodology described in WCAP-13749-P-A. The conditional exemption from the measurement improves plant availability and minimizes disruptions to normal plant operations. Plant safety criteria are not compromised by the conditional exemption of this one measurement.

<u>Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.1.3</u>

To address this issue, COL Appendix A Technical Specification SR 3.1.3.2 is proposed to be revised to add a second note stating "Not required to be performed provided applicable criteria in the COLR are satisfied" as a less restrictive change. In addition, Technical Specification 5.6.3, CORE OPERATING LIMITS REPORT (COLR), is revised to add references to 7a) WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004 (Westinghouse Proprietary) and WCAP-16045-NP-A, (Non-Proprietary), 7b) WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007 (Westinghouse Proprietary) and WCAP-16045-NP-A,

Addendum 1-A, (Non-Proprietary), and 7c) WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997 (Westinghouse Proprietary). Each of these three references are identified as Methodology for Specification 3.1.3 - Moderator Temperature Coefficient (MTC).

Currently, the COL Appendix A Technical Specifications require measurements of MTC during each fuel cycle near EOL to verify the most negative MTC lower limit. The measurement made near EOL is performed at or near HFP conditions. MTC measurements at HFP are more difficult to perform because of the small changes in the core average power during the measurement due to:

- small variations in soluble boron concentration,
- changes in xenon concentration and distribution,
- changes in fuel temperature, and
- changes in enthalpy rise.

Changes in each of these parameters must be accurately accounted for when reducing the measurement data, or additional measurement uncertainties will be introduced. Even though these additional uncertainties may be small, the total reactivity change associated with the swing in moderator temperature is also relatively small. The resulting MTC measurement uncertainty created by even a small change in power level can then become significant and, if improperly accounted for, can yield misleading measurement results.

Each measurement of MTC requires several hours at less than full power operation (as a buffer to measurement-induced transients) and requires additional operating staff resources. This presents a perturbation to normal operation and to the reactor itself. An alternate method is proposed to improve availability and minimize disruption to normal plant operations. The MTC measurement is replaced by a design calculation of the core MTC if predefined requirements are met. This is a less restrictive change.

The proposed change allows modification of the EOL MTC SR by placing a set of conditions on core operations. If these conditions are met (i.e., the specified revised prediction of the MTC and limits for several core parameters measured during the cycle are within specified bounds) the measurement of MTC would not be required. The conditional exemption from the HFP near EOL 300 ppm MTC measurement does not impact the safe operation of the plant. The safety analysis assumption of a constant moderator density coefficient and the actual value assumed does not change. Instead, a revised prediction of the MTC and limits for several core parameters measured during the cycle are verified to be within bounds specified in COLR. The method for calculating the revised prediction is consistent with the approved methodology of WCAP-13749-P-A.

The methodology for the proposed change was submitted to the NRC as Westinghouse Topical Report WCAP-13749 in May 1993. In October 1996, the NRC determined the report to be acceptable for referencing in license applications to the extent specified and

under the limitations stated in the Brookhaven technical evaluation report and the NRC staff's safety evaluation report. The topical report was approved by the NRC with two requirements:

- Only PHOENIX-P/ANC calculation methods described in WCAP-11596-P-A (Proprietary) and WCAP-11597-A (Nonproprietary), "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988, are used for the individual plant analyses relevant to determinations for the EOL MTC plant methodology, and

- the predictive correction is reexamined if changes in core fuel designs or continued MTC calculation/measurement data show significant effect on the predictive correction.

<u>Compliance with WCAP-13749-P-A Condition 1 – PHOENIX-P/ANC Calculation Methods</u>

As described in UFSAR Subsection 4.3.3.2, the AP1000 plant core design calculations are performed using the PARAGON lattice code using NEXUS methodology. The PARAGON lattice code is also capable of generating the group constants generated by PHOENIX-P, and has been benchmarked and qualified to the same degree as PHOENIX-P. The NRC has approved the use of PARAGON as an alternative method for generating all macroscopic and microscopic group constants for uranium fueled cores (WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004 (Westinghouse Proprietary) and WCAP-16045-NP-A, (Non-Proprietary)). The primary difference between PARAGON and PHOENIX-P is that PARAGON uses Collision Probability theory with the interface current method to solve the integral transport equation. PARAGON also allows increased flexibility in modeling the exact assembly and pin cell geometry. The group constants generated by PARAGON are coupled to the spatial few-group code using the NRC approved NEXUS nuclear data methodology (WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007 (Westinghouse Proprietary) and WCAP-16045-NP-A, Addendum 1-A, (Non-Proprietary)).

In Section 4.0, Conditions and Limitations of the NRC's Safety Evaluation (SE) for WCAP-16045-P-A, the NRC stated:

> "1. The PARAGON code can be used as a replacement for the PHOENIX-P lattice code, whenever the PHOENIX-P code is used in NRC approved methodologies."

The NEXUS methodology is a parameterization of the PARAGON nuclear data output for use within the ANC core simulator code to simplify the use of this code system for design use.

In Section 5.0, Conclusion, of the NRC's SE for WCAP-16045-P-A, Addendum 1-A, the NRC stated:

> "The NRC staff has reviewed the TR submitted by Westinghouse and determined that the NEXUS/ANC code system is adequate to replace the PARAGON/ANC code system wherever the latter is used in NRC approved methodologies."

As discussed above, the AP1000 plant core design calculations that are performed using the PARAGON/NEXUS/ANC system are equivalent to those performed with those using the PHOENIX/ANC system. Therefore, the use of PARAGON and NEXUS is consistent with condition (1) above in the NRC SER for WCAP-13749-P-A and satisfies the TER requirement to demonstrate the uncertainty limits assumed in WCAP-13749-P-A, as discussed on page 5 of the TER. The NRC used this TER as the basis for their SER.

Compliance with WCAP-13749-P-A Condition 2 – Effects on Predictive Correction

Prior to the use of the conditional exemption technique, SNC will confirm that core design changes and MTC calculation and measurement data are confirmed to not show a significant effect on the predictive correction. If a significant effect is found, the use of the predictive correction will be re-examined. All of the core performance benchmark criteria, which are confirmed from startup physics test results, from routine HFP boron concentration measurements, and from flux map surveillances performed during the cycle, must be met before the revised predicted MTC can be calculated per the prescribed algorithm in WCAP-13749-P-A.

Licensing Basis Precedent

Additional information on meeting these conditions in a similar manner as for the AP1000 plant is contained in the precedent licensing actions for the Southern Nuclear Operating Company (SNC) Joseph M. Farley Nuclear Plant Units 1 and 2, and Vogtle Electric Generating Plant Units 1 and 2, including the following:

- SNC Letter NL-14-0115, “Joseph M. Farley Nuclear Plant – Units 1 and 2, Vogtle Electric Generating Plant - Units 1 and 2, License Amendment Request to Revise Technical Specification Surveillance Requirement 3.1.3.2 and Specification 5.6.5,” by letter from SNC to the NRC, dated September 17, 2014 (ADAMS Accession No. ML14267A030), Enclosures 9 (proprietary) and 10 (non-proprietary).

- SNC Letter NL-15-0188, “Joseph M. Farley Nuclear Plant – Units 1 and 2, Vogtle Electric Generating Plant - Units 1 and 2, Response to Request for Additional Information Regarding the License Amendment Request to Revise Technical Specification Surveillance Requirement 3.1.3.2 and Specification 5.6.5,” by letter from SNC to the NRC, dated February 13, 2015 (ADAMS Accession No. ML15050A253).

- Joseph M. Farley, Units 1 and 2, and Vogtle Electric Generating Plant, Units 1 and 2, Amendment No.198 to Joseph M. Farley Nuclear Plant (Farley) Unit 1, Renewed Facility Operating License No. NPF-2, Amendment No.194 to Farley, Unit 2, Renewed Facility Operating License No. NPF-8, Amendment No. 174 to Vogtle Electric Generating Plant (VEGP), Unit 1, Renewed Facility Operating License NPF-68, and Amendment No. 156 to VEGP, Unit 2, Renewed Facility Operating License NPF-81, by letter from NRC to Southern Nuclear Operating Company, Inc., “Joseph M. Farley, Units 1 and 2, and Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments (TAC Nos. MF4828, MF4829, MF4889, and MF4890)” dated June 2, 2015 (ADAMS Accession No. ML15180A334).

Request to Eliminate Requirement to Submit a Most Negative Moderator Temperature Coefficient Limit Report

Although the NRC approved WCAP-13749-P-A is used as the basis for this license amendment request, an exception to reduce regulatory burden for both the NRC and the licensee is proposed. WCAP-13749-P-A requires submittal of a “Most Negative Moderator Temperature Coefficient Limit Report” to the NRC. It is proposed that the report not be required to be submitted. The report serves no apparent technical or business need. The applicability restrictions in the WCAP, the algorithm, and the acceptance criteria of the proposed report would be included in the station procedure governing the EOL MTC surveillance. There is no compelling reason that this particular surveillance should require notifying the NRC prior to performing the surveillance procedure.

The exception of not including a "Most Negative Moderator Temperature Coefficient Limit Report" that is contained in WCAP-13749-P-A was approved by the NRC in the licensing basis precedent for Joseph M. Farley Nuclear Plant Units 1 and 2, and Vogtle Electric Generating Plant Units 1 and 2 previously referenced.

Description of any Changes to Current Licensing Basis Documents

COL Appendix A Technical Specifications Changes:

The following changes to the COL Appendix A Technical Specifications are proposed:

1. Technical Specification 3.1.3 is revised as follows:

    a. SURVEILLANCE REQUIREMENT SR 3.1.3.2 is revised to add a new NOTE 1 stating “Not required to be performed provided applicable criteria in the COLR are satisfied” for providing for the allowance to replace the requirement to measure the MTC after reaching the equivalent of an equilibrium RTP ARO boron concentration of 300 ppm near EOL, with a revised prediction of MTC at EOL using an algorithm from WCAP-13749-P-A.

    b. Existing NOTE 1 is moved and renumbered as NOTE 2.

2. Technical Specification 5.6.3 is revised to add a new Reference 7a) as WCAP-16045-P-A, “Qualification of the Two-Dimensional Transport Code PARAGON,” August 2004 (Westinghouse Proprietary) and WCAP-16045-NP-A, (Non-Proprietary), 7b) as WCAP-16045-P-A, Addendum 1-A, “Qualification of the NEXUS Nuclear Data Methodology,” August 2007 (Westinghouse Proprietary) and WCAP-16045-NP-A, Addendum 1-A, (Non-Proprietary), and 7c) as “WCAP-13749-P-A, “Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement,” March 1997 (Westinghouse Proprietary). Each of these three references are identified with the following: “(Methodology for Specification 3.1.3 - Moderator Temperature Coefficient (MTC).)”

UFSAR Changes:

The following licensing basis changes to the UFSAR are proposed:

None.

2.5 Changes to COL Appendix A Technical Specification 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z))

This activity affects the means for maintaining power distributions within the required absolute power generation limits as described in COL Appendix A Technical Specification 3.2.1. The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core. $F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution. With the OPDMS monitoring parameters, peak linear heat rate (which is proportional to $F_Q(Z)$) is measured continuously. With the OPDMS not monitoring parameters, $F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

The OPDMS provides the operator with the current allowable operating space, detailed current power distribution information, thermal margin assessment and operational recommendations to manage and maintain required thermal margins. As such, the OPDMS provides the primary means of managing and maintaining required operating thermal margins during normal operation.

In the unlikely event that the OPDMS is out of service, power distribution controls based on bounding, precalculated analysis are also provided to the operator such that the OPDMS is not a required element for short-term reactor operation. Limits are placed on the AFD so that the heat flux hot channel factor $F_Q(Z)$ is maintained within acceptable limits. The axial power distribution procedures are part of the required operating procedures followed in normal operation with the OPDMS out of service. In service, the OPDMS provides continuous indication of power distribution, shutdown margin, and margin to design limits.

COL Appendix A Technical Specification 3.2.1 requires changes to address the following issues:

- The Applicability does not reflect that the fixed incore detectors cannot function adequately to measure $F_Q(Z)$ at low power levels below approximately 20% RTP. Therefore, based on the current Applicability of MODE 1, the LCO would be applicable up to approximately 20% RTP with no mechanism for performing the required SRs.

- The Note modifying SR 3.2.1.3 inappropriately allows 31 days of operation with the OPDMS not functional (i.e., not monitoring parameters) before the initial verification that $F_Q^C(Z)$ remains within limit as specified in the COLR is required to be performed.

- The Note modifying SR 3.2.1.4 inappropriately allows 31 days of operation with the OPDMS not functional (i.e., not monitoring parameters) before the initial verification that $F_Q^W(Z)$ remains within limits as specified in the COLR is required to be performed.

Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.2.1

To address this issue, COL Appendix A Technical Specification 3.2.1 is proposed to be revised as follows:

- The Applicability is proposed to be changed from "MODE 1" to "MODE 1 with THERMAL POWER ≥ 25% RTP and" and the end state for Required Action C.1 for exiting the Applicability is proposed to be changed from "Be in MODE 2" to "Reduce THERMAL POWER to < 25% RTP" for consistency with the proposed Applicability. This is a less restrictive change. Establishing applicability as MODE 1 with THERMAL POWER ≥ 25% RTP allows for functional fixed incore detectors to be available to measure $F_Q(Z)$ to verify that this parameter is within the limits specified in the COLR. Applicability in MODE 1 with THERMAL POWER < 25% RTP is not required because local power peaking at very low power levels is not limiting with respect to design basis accident analyses and because the incore instrumentation system may not be able to provide an accurate measurement of the core flux distribution at very low THERMAL POWER levels.

- The Note modifying SR 3.2.1.3 is proposed to be revised to require the initial verification that $F_Q^C(Z)$ remains within limit as specified in the COLR to be performed within 24 hours after the OPDMS is declared not functional (i.e., not monitoring parameters) as a more restrictive change. Subsequent surveillance remains required 31 EFPD thereafter. This change is acceptable as it provides enough time to perform the surveillance while verifying fuel design limits remain protected while the OPDMS is not functional.

- The Note modifying SR 3.2.1.4 is proposed to be revised to require the initial verification that $F_Q^W(Z)$ remains within limits as specified in the COLR to be performed within 24 hours after the OPDMS is declared not functional (i.e., not monitoring parameters) as a more restrictive change. Subsequent surveillance remains required 31 EFPD thereafter. This change is acceptable as it provides enough time to perform the surveillance while verifying fuel design limits remain protected while the OPDMS is not functional.

Description of any Changes to Current Licensing Basis Documents

COL Appendix A Technical Specifications Changes:

The following changes to the COL Appendix A Technical Specifications are proposed:

1. Technical Specification 3.2.1 is revised as follows:

a. The APPLICABILITY is revised to state: "MODE 1 with THERMAL POWER ≥ 25% RTP and with On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters."

b. REQUIRED ACTION C.1 is revised to delete "Be in MODE 2" and replace with "Reduce THERMAL POWER to < 25% RTP."

c. SURVEILLANCE REQUIREMENTS SR 3.2.1.3 NOTE is revised to state: "Not required to be performed until 24 hours after OPDMS not monitoring parameters."

d. SURVEILLANCE REQUIREMENTS SR 3.2.1.4 NOTE 1 is revised to state: "Not required to be performed until 24 hours after OPDMS not monitoring parameters."

UFSAR Changes:

The following licensing basis changes to the UFSAR are proposed:

None.

2.6 Changes to COL Appendix A Technical Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

This activity affects the means for maintaining power distributions within the required absolute power generation limits as described in COL Appendix A Technical Specification 3.2.2. The purpose of the limits on the power density at any point in the core is to ensure that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors assures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. $F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup. With the OPDMS monitoring parameters, $F^N_{\Delta H}$ is determined continuously by the OPDMS. When the OPDMS is not monitoring parameters, $F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed to determine $F^N_{\Delta H}$.

The COLR provides peaking factor limits that ensure that the design basis value of the DNB is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio. Transient events that may be DNB limited are assumed to begin with a $F^N_{\Delta H}$ that satisfies the COL Appendix A Technical Specification 3.2.2 LCO requirements. Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is

no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

For transients that may be DNB limited, the reactor coolant system flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion. This value provides a high degree of assurance that the hottest fuel rod in the core will not experience a DNB. The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of COL Appendix A Technical Specification 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation. The loss-of-coolant accident (LOCA) safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature.

The OPDMS provides the operator with the current allowable operating space, detailed current power distribution information, thermal margin assessment and operational recommendations to manage and maintain required thermal margins. As such, the OPDMS provides the primary means of managing and maintaining required operating thermal margins during normal operation.

In the unlikely event that the OPDMS is out of service, power distribution controls based on bounding, precalculated analysis are also provided to the operator such that the online monitoring system is not a required element for short-term reactor operation. The axial power distribution procedures are part of the required operating procedures followed in normal operation with the OPDMS out of service. In service, the OPDMS provides continuous indication of power distribution, shutdown margin, and margin to design limits.

COL Appendix A Technical Specification 3.2.2 requires changes to address the following issues:

- The Applicability does not reflect that the fixed incore detectors cannot function adequately to measure $F^N_{\Delta H}$ at low power levels below approximately 20% RTP. Therefore, based on the current Applicability of MODE 1, the LCO would be applicable up to approximately 20% RTP with no mechanism for performing the required SRs.

- The Note modifying SR 3.2.2.2 inappropriately allows 31 days of operation with the OPDMS not functional (i.e., not monitoring parameters) before the initial verification that $F^N_{\Delta H}$ remains within limits as specified in the COLR is required to be performed.

Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.2.2

To address this issue, COL Appendix A Technical Specification 3.2.2 is proposed to be revised as follows:

- The Applicability is proposed to be changed from "MODE 1" to "MODE 1 with THERMAL POWER ≥ 25% RTP and" and the end state for Required Action B.1 for exiting the Applicability is proposed to be changed from "Be in MODE 2" to "Reduce THERMAL POWER to < 25% RTP" for consistency with the proposed Applicability. This is a less restrictive change. Establishing applicability as MODE 1 with THERMAL POWER ≥ 25% RTP allows for functional fixed incore detectors to be available to measure $F^N_{\Delta H}$ to verify that this parameter is within the limits specified in the COLR. Applicability in MODE 1 with THERMAL POWER < 25% RTP is not required because local power peaking at very low power levels is not limiting with respect to design basis accident analyses and because the incore instrumentation system may not be able to provide an accurate measurement of the core flux distribution at very low THERMAL POWER levels.

- The Note modifying SR 3.2.2.2 is proposed to be revised to require the initial verification that $F^N_{\Delta H}$ remains within limit as specified in the COLR to be performed within 24 hours after the OPDMS is declared not functional (i.e., not monitoring parameters) as a more restrictive change. Subsequent surveillance remains required 31 EFPD thereafter. This change is acceptable as it provides enough time to perform the surveillance while verifying fuel design limits remain protected while the OPDMS is not functional.

Description of any Changes to Current Licensing Basis Documents

COL Appendix A Technical Specifications Changes:

The following changes to the COL Appendix A Technical Specifications are proposed:

1. Technical Specification 3.2.2 is revised as follows:

    a. The APPLICABILITY is revised to state: "MODE 1 with THERMAL POWER ≥ 25% RTP and with On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters."

    b. REQUIRED ACTION B.1 is revised to delete "Be in MODE 2" and replace with "Reduce THERMAL POWER to < 25% RTP."

    c. SURVEILLANCE REQUIREMENTS SR 3.2.2.2 NOTE is revised to state: "Not required to be performed until 24 hours after OPDMS not monitoring parameters."

UFSAR Changes:

The following licensing basis changes to the UFSAR are proposed:

None.

2.7 Changes to COL Appendix A Technical Specification 3.2.5, On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters

This activity affects the means for maintaining power distributions within the required absolute power generation limits as described in COL Appendix A Technical Specification 3.2.5. The purpose of the limits on the OPDMS-monitored power distribution parameters is to provide assurance of fuel integrity during Conditions I (Normal Operation) and II (incidents of Moderate Frequency) events by: (1) not exceeding the minimum DNBR in the core, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the peak cladding temperature (PCT) limit of 2200°F is not exceeded.

Limits on linear power density or peak kw/ft assure that the peak linear heat rate assumed as a base condition in the LOCA analyses is not exceeded during normal operation. Limits on $F^N_{\Delta H}$ ensure that the LOCA analysis assumptions and assumptions made with respect to the overtemperature ΔT setpoint are maintained. The limit on DNBR ensures that if transients analyzed in the safety analyses initiate from the conditions within the limit allowed by the OPDMS, the DNB criteria will be met.

The OPDMS provides the operator with the current allowable operating space, detailed current power distribution information, thermal margin assessment and operational recommendations to manage and maintain required thermal margins. As such, the OPDMS provides the primary means of managing and maintaining required operating thermal margins during normal operation.

In the unlikely event that the OPDMS is out of service, power distribution controls based on bounding, precalculated analysis are also provided to the operator such that the online monitoring system is not a required element for short-term reactor operation. The axial power distribution procedures are part of the required operating procedures followed in normal operation with the OPDMS out of service. In service, the OPDMS provides continuous indication of power distribution, shutdown margin, and margin to design limits.

<u>Description of Rapid Power Reduction System</u>

As described in UFSAR Subsections 4.3.2.4.18, 7.7.1, and 7.7.1.10, the rapid power reduction system is designed with the capability of responding to large, rapid load rejections (turbine trip or grid disconnect from 50% RTP reactor power or greater) or the loss of a main feedwater pump train without initiating a reactor trip using the normal rod control system, reactor control system, and the rapid power reduction system. This results in a reduction of thermal power to a level that can be handled by the steam dump system sized to pass 40% of nominal steam flow in conjunction with the reactor power control system for the large load rejection. For a loss of main feedwater pump transient, the actuation of the rapid power reduction reduces power to a level that can be handled by the remaining feedwater pumps. Load rejections requiring greater than a fifty percent reduction

of rated thermal power or the loss of a main feedwater pump train as indicated by the pump motor breaker positions initiates the rapid power reduction system when above a predetermined power level. The rapid power reduction system utilizes selected (either automatically or manually) control rod groups and/or banks which are intentionally tripped to rapidly reduce reactor power into a range where the rod control and reactor control systems are sufficient to maintain stable plant operation. The number of rods needed to obtain the required power reduction is dependent on the initiating transient and the transient initial conditions, including the core burnup during the fuel cycle, the initial power level, and the initial position of the MSHIM gray and control rod banks. Therefore, bank selection logic is provided to automatically determine which rods will be released by the rapid power reduction system based on these factors. The consequences of accidental or inappropriate actuation of the rapid power reduction system are included in the cycle specific safety analysis and licensing process.

COL Appendix A Technical Specification 3.2.5 requires changes to address the following issues:

- The LCO specifying the parameters that shall not exceed their operating limits as specified in the COLR include “Peak Linear Power Density.” This terminology is not used in the AP1000 plant calculations and topical reports, including the COLR, for this parameter.

- The Applicability with OPDMS monitoring parameters a, b, and c of “MODE 1 with THERMAL POWER > 50% RTP” was based on an assumption that the rapid power reduction system would be blocked if the OPDMS was not functioning. The rapid power reduction system, and secondary-side steam dump capability, is designed to prevent reactor trip and lifting of steam generator safety valves following a step decrease in turbine load from full power (greater than 50% turbine power reduction at a rapid rate) that results in a rapid mismatch between nuclear and turbine power. This is accomplished upon the detection of a large and rapid turbine power reduction (via a rate/lag circuit, similar to that used for steam dump control), by a circuit that provides a signal demanding the release of a selected number of control rods when power is above a predetermined level. This assumption that the rapid power reduction system would be blocked if the OPDMS is not functioning is no longer required as a design commitment. This results in the possibility that fuel design limits can be exceeded following rapid power reduction system actuation at power levels < 50% RTP but ≥ 25% RTP if the OPDMS monitored parameters a, b, and c are not being monitored and exceed the limits for each as specified in the COLR.

<u>Technical Evaluation of Proposed Changes to COL Appendix A Technical Specification 3.2.5</u>

To address this issue, COL Appendix A Technical Specification 3.2.5 is proposed to be revised as follows:

- The LCO specifying the parameters that shall not exceed their operating limits as specified in the COLR is proposed to be revised to change “Peak Linear Power Density” to “Peak Linear Heat Rate” as an administrative change to reflect the terminology used

in the AP1000 plant calculations and topical reports, including the COLR, for this parameter.

- The Applicability of "MODE 1 with THERMAL POWER > 50% RTP with OPDMS monitoring parameters a, b, and c" is proposed to be revised to "MODE 1 with THERMAL POWER ≥ 25% RTP and with OPDMS monitoring parameters a, b, and c," and the end state for Required Action B.1 for exiting the Applicability is proposed to be changed from "Reduce THERMAL POWER to ≤ 50% RTP" to "Reduce THERMAL POWER to < 25% RTP" for consistency with the proposed Applicability, as a more restrictive change. This addresses the possibility that fuel design limits can be exceeded following rapid power reduction system actuation at power levels < 50% RTP but ≥ 25% RTP if the OPDMS monitored parameters a, b, and c are not being monitored and exceed the limits for each as specified in the COLR. This change is acceptable to prevent exceeding fuel design limits during design basis events in MODE 1 with THERMAL POWER ≥ 25% RTP.

Description of any Changes to Current Licensing Basis Documents

COL Appendix A Technical Specifications Changes:

The following changes to the COL Appendix A Technical Specifications are proposed:

1. Technical Specification 3.2.5 is revised as follows:

   a. LCO 3.2.5.a is revised to replace "Peak Linear Power Density" with "Peak Linear Heat Rate."

   b. The APPLICABILITY is revised to state: "MODE 1 with THERMAL POWER ≥ 25% RTP and with OPDMS monitoring parameters a, b, and c."

   c. REQUIRED ACTION B.1 is revised to replace "≤ 50% RTP" with "< 25% RTP."

UFSAR Changes:

The following licensing basis changes to the UFSAR are proposed:

None.

2.8 Technical Evaluation of Other Impacts

An impact review determined that these proposed changes would have negligible impact on the AP1000 plant PRA presented in UFSAR Chapter 19, including the Fire PRA, results and insights (e.g., core damage frequency (CDF) and large release frequency (LRF)). The proposed changes to the design of the PMS automatic reactor trips and the crediting of PMS automatic reactor trips necessary to prevent exceeding the fuel design limits including the power range high neutron flux (high setpoint) trip, the power range high positive flux rate trip, the overpower ΔT trip, and the overtemperature ΔT trip; and changes to the COL Appendix A Technical Specifications for maintaining MTC within the required reactivity control limits and maintaining power generation within the required power distribution limits,

do not impact any initiating event and do not introduce any new failure modes or mechanisms. There is no impact to or addition of any structure, system, component (SSC) that is considered to be D-RAP risk significant (DCD Tier 1 Table 3.7-1 and UFSAR Table 17.4-1). There is no interface with the diverse actuation system (DAS), and no change to the design functions of the DAS to provide diverse reactor protection system functions.

The proposed PMS changes do not adversely affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. No safety-related SSC or function is adversely involved. The proposed changes do not involve nor interface with any SSC accident initiator or initiating sequence of events related to the accidents evaluated in the UFSAR. The proposed changes do not adversely affect the ability of the PMS automatic reactor trips to perform the required safety function to trip the reactor when necessary to protect fuel design limits, and do not adversely affect the probability of inadvertent operation or failure of the PMS automatic reactor trips. The proposed changes to the methods for maintaining MTC within the required reactivity control limits and maintaining power generation within the required power distribution limits do not result in any increase in probability of an analyzed accident occurring, and prevent power oscillations and maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects. The proposed changes do not affect the radiological source terms (i.e., amounts and types of radioactive materials released, their release rates and release durations) used in the accident analyses.

The proposed changes do not require a change to procedures or method of control that adversely affects the performance of the PMS safety-related design functions as described in the UFSAR. The physical operation of the PMS, including as-installed inspections, testing, and maintenance requirements, as described in the UFSAR are not changed, with the exception of the software changes to implement the revised overpower ΔT and overtemperature ΔT trip setpoints and $f(\Delta I)$ penalty shapes. Therefore, there are no changes to procedures or a method of control that adversely impact the licensing basis. The proposed changes maintain the design functions of the PMS to be available to mitigate the required transient and accident conditions, and to prevent exceeding the fuel design limits.

The proposed changes do not adversely interface with or adversely affect safety-related equipment or a fission product barrier. The proposed changes to the PMS automatic reactor trips required to trip the reactor when necessary to protect fuel design limits are designed and comply with the regulatory requirements described in the UFSAR. The proposed changes do not result in a new failure mode, malfunction or sequence of events that could adversely affect a radioactive material barrier or safety-related equipment. The proposed changes do not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that would result in significant fuel cladding failures.

The proposed changes do not adversely affect safety-related equipment or equipment whose failure could initiate an accident. The proposed changes to the PMS automatic

reactor trips required to trip the reactor when necessary to protect fuel design limits do not adversely interface with or adversely affect safety-related equipment or a radioactive material barrier. The proposed changes do not adversely affect any safety-related equipment, design code limit allowable value, safety-related function or design analysis, nor do they adversely affect any safety analysis input or result, or design/safety margin. Instead, the changes proposed preserve required design/safety margins.

The Technical Specification Safety Limits are not affected. The Limiting Safety System Settings, Limiting Control Settings, and Limiting Conditions for Operation requirements continue to be met by the proposed changes to the PMS automatic reactor trips required to trip the reactor when necessary to protect fuel design limits, and by the proposed changes to the methods for maintaining MTC within the required reactivity control limits and maintaining power generation within the required power distribution limits, so that affected safety system functions are met and maintained operable. The respective Technical Specifications Bases are revised under the Technical Specifications Bases Control Program to support the identified COL Appendix A Technical Specifications changes and additional changes to the UFSAR, and are provided for information only.

There are no radiation zone changes or radiological access control changes required because of these proposed changes. The physical design and operation of the PMS as described in the UFSAR that may affect the radiation protection requirements are not changed, and thus there are no changes required to the radiation protection design features described in UFSAR Section 12.3.

There are no fire area changes required because of these proposed changes. The proposed changes do not require any changes to the fire protection analysis described in UFSAR Appendix 9A.

There is no change to the risk significant designation of SSCs within the Design Reliability Assurance Program as described in UFSAR Table 17.4-1.

The proposed changes do not affect the containment, control, channeling, monitoring, processing or releasing of radioactive and non-radioactive materials. The proposed changes maintain the design functions of the PMS to be available to mitigate the required transient and accident conditions, and to prevent exceeding the fuel design limits. No effluent release path is affected. The types and quantities of expected effluents are not changed. Therefore, radioactive or non-radioactive material effluents are not affected.

The proposed changes do not affect plant radiation zones, controls under 10 CFR 20, and expected amounts and types of radioactive materials, as the physical design changes proposed for the PMS do not result in any changes to these radiological conclusions as described in the UFSAR. Therefore, individual and cumulative radiation exposures do not change.

The proposed changes do not affect the results of the aircraft impact assessment described in UFSAR Subsection 19F.4.

The proposed changes have no adverse impact on the emergency plan or the physical security plan implementation, because there are no changes to physical access to credited equipment inside the Nuclear Island (including containment or the auxiliary building) and no adverse impact to plant personnel's ability to respond to any plant operations or security event.

2.9 Summary

The proposed changes would revise COL Appendix A Technical Specifications information, and associated UFSAR information, concerning the crediting of PMS automatic reactor trips necessary to prevent exceeding the fuel design limits including the power range high neutron flux (high setpoint) trip, the power range high positive flux rate trip, the overpower ΔT trip, and the overtemperature ΔT trip; and changes to the COL Appendix A Technical Specifications for maintaining MTC within the required reactivity control limits and maintaining power generation within the required power distribution limits. The proposed changes do not adversely affect the design functions of the PMS.

The proposed changes do not adversely affect any safety-related equipment or function, design function, radioactive material barrier, or safety analysis.

3. TECHNICAL EVALUATION (Included in Section 2)

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 52.98(c) requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a COL. The proposed changes involve a change to COL Appendix A Technical Specifications information. Therefore, NRC approval is required prior to making the plant-specific proposed changes in this license amendment request.

10 CFR 52, Appendix D, Section VIII.B.5.a allows an applicant or licensee who references this appendix to depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of the section. The proposed changes for the PMS, which include changes to the UFSAR, involve a revision to COL Appendix A Technical Specifications information. Therefore, NRC approval is required for the Tier 2 and the Technical Specification changes.

10 CFR 52, Appendix D, VIII.C.6 states that after issuance of a license, "Changes to the plant-specific TS (Technical Specifications) will be treated as license amendments under 10 CFR 50.90." 10 CFR 50.90 addresses the applications for amendments of licenses, construction permits and early site permits. As discussed above, changes to Technical Specifications are requested. Therefore, NRC approval is required for these Technical Specification changes.

10 CFR 50.36 Technical specifications. – (c) *Technical specifications will include items in the following categories: (1) Safety limits, limiting safety system settings, and limiting control settings.* (*2) Limiting conditions for operation. (3) Surveillance Requirements.* The safety limits are not affected. In addition, except where justified by this license amendment request, the limiting safety system settings, limiting control settings, and limiting conditions for operation requirements continue to be met with the proposed changes to the PMS including reactor trip system instrumentation, and with the proposed changes to the reactivity control systems and power distribution limits requirements. The PMS and reactor trip system instrumentation safety functions are met by these proposed changes, and are maintained operable by the proposed COL Appendix A Technical Specification changes. Changes proposed to the Surveillance Requirements for the reactor trip system instrumentation, reactivity control systems, and power distribution limits maintain the initial conditions and operating limits required by the accident analysis. The associated Technical Specifications Bases are revised under the Technical Specifications Bases Control Program to support these COL Appendix A Technical Specifications changes and additional changes to the UFSAR, and are provided for information only.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2 requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena, such as earthquakes. The proposed changes to the PMS including reactor trip system instrumentation are designed to the existing seismic design requirements, including seismic Category I requirements. The proposed changes do not involve physical modifications or addition of systems, structures, and components, except for software changes for PMS reactor trip system instrumentation, and do not impact the existing seismic design requirements for the reactor or reactor control systems, including the rod control system. Therefore, the proposed changes comply with the requirements of GDC 2.

10 CFR Part 50, Appendix A, GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Changes proposed for the PMS reactor trip system instrumentation, reactivity control systems, and power distribution limits maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects. Therefore, the proposed changes comply with the requirements of GDC 10.

10 CFR Part 50, Appendix A, GDC 11 requires that the reactor core and associated coolant systems be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. Changes proposed for the PMS reactor trip system instrumentation, reactivity control systems, and power distribution limits maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that the acceptance criteria for events resulting in

positive reactivity insertion and reactivity feedback effects are met. Therefore, the proposed changes comply with the requirements of GDC 11.

10 CFR Part 50, Appendix A, GDC 12 requires a core design to assure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible. Changes proposed for the PMS reactor trip system instrumentation, reactivity control systems, and power distribution limits prevent power oscillations and maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects. Therefore, the proposed changes comply with the requirements of GDC 12.

4.2 Precedent

4.2.1 Changes to COL Appendix A Technical Specification 3.1.3, Moderator Temperature Coefficient (MTC)

The changes to COL Appendix A Technical Specification 3.1.3 to replace the requirement to measure the MTC after reaching the equivalent of an equilibrium RTP (also described as HFP in WCAP-13749-P-A) ARO boron concentration of 300 ppm near EOL, with a revised prediction of MTC at EOL using an algorithm from WCAP-13749-P-A, was previously approved for numerous Westinghouse reactors. This includes the following license amendment requests, responses to NRC requests for additional information (RAIs), and issued license amendments for the Southern Nuclear Operating Company (SNC) Joseph M. Farley Nuclear Plant Units 1 and 2, and SNC Vogtle Electric Generating Plant Units 1 and 2:

- SNC Letter NL-14-0115, “Joseph M. Farley Nuclear Plant – Units 1 and 2, Vogtle Electric Generating Plant - Units 1 and 2, License Amendment Request to Revise Technical Specification Surveillance Requirement 3.1.3.2 and Specification 5.6.5,” by letter from SNC to the NRC, dated September 17, 2014 (ADAMS Accession No. ML14267A030).

- SNC Letter NL-15-0188, “Joseph M. Farley Nuclear Plant – Units 1 and 2, Vogtle Electric Generating Plant - Units 1 and 2, Response to Request for Additional Information Regarding the License Amendment Request to Revise Technical Specification Surveillance Requirement 3.1.3.2 and Specification 5.6.5,” by letter from SNC to the NRC, dated February 13, 2015 (ADAMS Accession No. ML15050A253).

- Joseph M. Farley, Units 1 and 2, and Vogtle Electric Generating Plant, Units 1 and 2, Amendment No.198 to Joseph M. Farley Nuclear Plant (Farley) Unit 1, Renewed Facility Operating License No. NPF-2, Amendment No.194 to Farley, Unit 2, Renewed Facility Operating License No. NPF-8, Amendment No. 174 to Vogtle Electric Generating Plant (VEGP), Unit 1, Renewed Facility Operating License NPF-68, and Amendment No. 156 to VEGP, Unit 2, Renewed Facility Operating License NPF-81, by letter from NRC to Southern Nuclear Operating Company, Inc., “Joseph M. Farley, Units 1 and 2, and Vogtle Electric Generating Plant, Units 1 and 2, Issuance of

Amendments (TAC Nos. MF4828, MF4829, MF4889, and MF4890)" dated June 2, 2015 (ADAMS Accession No. ML15180A334).

4.3 Significant Hazards Consideration

The proposed changes would revise the Combined License (COL) in regards to detailed design of the protection and safety monitoring system (PMS) automatic reactor trips and the crediting of PMS automatic reactor trips necessary to prevent exceeding the fuel design limits including the power range high neutron flux (high setpoint) trip, the power range high positive flux rate trip, the overpower ΔT trip, and the overtemperature ΔT trip; and changes to the COL Appendix A Technical Specifications for maintaining moderator temperature coefficient within the required reactivity control limits and maintaining power generation within the required power distribution limits.

The requested amendment proposes changes to COL Appendix A Technical Specifications and Updated Final Safety Analysis Report (UFSAR) Tier 2 information.

An evaluation to determine whether a significant hazards consideration is involved with the requested amendment was completed by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No

The proposed changes do not adversely affect the operation of any systems or equipment that initiate an analyzed accident or alter any structures, systems, and components (SSCs) accident initiator or initiating sequence of events. The proposed changes do not adversely affect the ability of the PMS automatic reactor trips to perform the required safety function to trip the reactor when necessary to protect fuel design limits, and do not adversely affect the probability of inadvertent operation or failure of the PMS automatic reactor trips. The proposed changes to the methods for maintaining moderator temperature coefficient within the required reactivity control limits and maintaining power generation within the required power distribution limits do not result in any increase in probability of an analyzed accident occurring, and prevent power oscillations and maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects.

Therefore, the requested amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed changes do not affect the operation of any systems or equipment that may initiate a new or different kind of accident, or alter any SSC such that a new accident initiator or initiating sequence of events is created. The proposed changes do not adversely affect the ability of the PMS automatic reactor trips to perform the required safety function to trip the reactor when necessary to protect fuel design limits, and do not adversely affect the probability of inadvertent operation or failure of the PMS automatic reactor trips. The proposed changes to the methods for maintaining moderator temperature coefficient within the required reactivity control limits and maintaining power generation within the required power distribution limits do not result in the possibility of an accident occurring, and prevent power oscillations and maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects.

These proposed changes do not adversely affect any other SSC design functions or methods of operation in a manner that results in a new failure mode, malfunction, or sequence of events that affect safety-related or nonsafety-related equipment. Therefore, this activity does not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that results in significant fuel cladding failures.

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

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

Response: No

The proposed changes maintain existing safety margins. The proposed changes to the PMS reactor trip system instrumentation, reactivity control systems, and power distribution limits maintain existing safety margin through continued application of the existing requirements of the UFSAR. The proposed changes maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that the existing fuel design limits specified in the UFSAR are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects. Therefore, the proposed changes satisfy the same safety functions in accordance with the same requirements as stated in the UFSAR. These changes do not adversely affect any design code, function, design analysis, safety analysis input or result, or design/safety margin.

No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed changes, and no margin of safety is reduced. Therefore, the requested amendment does not involve a significant reduction in a margin of safety.

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. Therefore, it is concluded that the requested 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.

5. ENVIRONMENTAL CONSIDERATIONS

The requested amendment requires changes to the Combined License (COL) in regards to detailed design of the protection and safety monitoring system (PMS) automatic reactor trips and the crediting of PMS automatic reactor trips necessary to prevent exceeding the fuel design limits including the power range high neutron flux (high setpoint) trip, the power range high positive flux rate trip, the overpower ΔT trip, and the overtemperature ΔT trip; and changes to the COL Appendix A Technical Specifications for maintaining moderator temperature coefficient within the required reactivity control limits and maintaining power generation within the required power distribution limits.

The requested amendment proposes changes to COL Appendix A Technical Specifications and Updated Final Safety Analysis Report (UFSAR) Tier 2 information.

A review has determined that the anticipated effects on facility construction and operation following implementation of the requested amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), in that:

(i) There is no significant hazards consideration.

As documented in Section 4.3, Significant Hazards Consideration, of this license amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment." The Significant Hazards Consideration determined that (1) the requested amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the requested amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the requested amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the requested 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.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed changes are unrelated to any aspect of plant construction or operation that would introduce any change to effluent types (e.g., effluents containing chemicals or biocides, sanitary system effluents, and other effluents), or affect any plant radiological or non-radiological effluent release quantities. Furthermore, the proposed changes do not affect any effluent release path or diminish the design functions or operational features that are credited with controlling the release of effluents during plant operation. Therefore, it is concluded that the requested amendment does not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes do not adversely affect walls, floors, or other structures that provide shielding. Plant radiation zones are not affected, and there are no changes to the controls required under 10 CFR Part 20 that preclude a significant increase in occupational radiation exposure. Therefore, the requested amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the requested amendment, it has been determined that anticipated construction and operational impacts of the requested amendment do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the requested 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 requested amendment.

6. REFERENCES

None.

Southern Nuclear Operating Company

ND-17-1495

Enclosure 2

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Proposed Changes to the Licensing Basis Documents

(LAR-17-031)

Additions identified by blue underlined text.
Deletions identified by red strikethrough of text.
* * * indicates omitted existing text that is not shown.

(This Enclosure consists of 54 pages, including this cover page.)

COL Appendix A, Technical Specification 3.1.3, Moderator Temperature Coefficient (MTC), is revised as follows:

1. Technical Specification 3.1.3 ACTIONS are revised as follows:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within upper limit.	A.1 ~~Establish administrative withdrawal limits for control banks to maintain~~ Restore MTC within limit.	24 hours

2. Technical Specification 3.1.3 SURVEILLANCE REQUIREMENTS are revised as follows:

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	---------- **- NOTES -** 1. Not required to be performed provided applicable criteria in the COLR are satisfied. ~~1~~2. Not required to be performed if the MTC measured at the equivalent of equilibrium RTP all rods out (ARO) boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. ---------- Verify MTC is within lower limit.	Once within 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP ARO boron concentration of 300 ppm AND 14 EFPD thereafter when MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR

COL Appendix A, Technical Specification 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z)), is revised as follows:

1. Technical Specification 3.2.1 APPLICABILITY is revised as follows:

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 25% RTP and with On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

2. Technical Specification 3.2.1 ACTIONS are revised as follows:

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------- **- NOTE -** Required Action A.~~4~~3 shall be completed whenever this Condition is entered. ---------- $F_Q^C(Z)$ not within limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	~~AND~~		
	~~A.2~~	~~Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.~~	~~72 hours after each $F_Q^C(Z)$ determination~~
	AND		
	A.~~3~~2	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND		
	A.~~4~~3	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

3. Technical Specification 3.2.1 ACTIONS are revised as follows:

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ---------- **- NOTE -** Required Action B.~~4~~3 shall be completed whenever this Condition is entered. ---------- $F_Q^W(Z)$ not within limits.	B.1	Reduce THERMAL POWER ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	~~AND~~		
	~~B.2~~	~~Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.~~	~~72 hours~~
	AND		
	B.~~3~~2	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	AND		
	B.~~4~~3	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.1

4. Technical Specification 3.2.1 ACTIONS are revised as follows:

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1	~~Be in MODE 2~~Reduce THERMAL POWER to < 25% RTP.	6 hours

5. Technical Specification 3.2.1 SURVEILLANCE REQUIREMENTS are revised as follows:

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	---------- **- NOTE -** Not required to be performed until ~~31 days~~24 hours after ~~the last verification of~~ OPDMS not monitoring parameters. ---------- Verify $F_Q^C(Z)$ within limit.	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified AND 31 effective full power days (EFPD) thereafter

6. Technical Specification 3.2.1 SURVEILLANCE REQUIREMENTS are revised as follows:

SURVEILLANCE		FREQUENCY
SR 3.2.1.4	---------- **- NOTES -** 1. Not required to be performed until ~~31 days~~24 hours after ~~the last verification of~~ OPDMS not monitoring parameters. 2. If $F_Q^W(Z)$ measurements indicate maximum over z $F_Q^W(Z)$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.4 once per 7 EFPD until two successive flux maps indicate maximum over z $F_Q^C(Z)$ has not increased. ----------	
	Verify $F_Q^W(Z)$ within limits.	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

COL Appendix A, Technical Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), is revised as follows:

1. Technical Specification 3.2.2 APPLICABILITY is revised as follows:

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 25% RTP and with On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

2. Technical Specification 3.2.2 ACTIONS are revised as follows:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------- **- NOTE -** Required Actions A.2 and A.3 shall be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limit.	A.1.1 Restore $F^N_{\Delta H}$ to within limit.	4 hours
	OR	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
	A.1.2.2 Reduce ~~Power Range Neutron Flux – High~~Overpower ΔT trip setpoints to ≤ 55% RTP.	72 hours
	AND	
	A.2 Perform SR 3.2.2.1.	24 hours
	AND	
	A.3 ---------- **- NOTE -** THERMAL POWER does not have to be reduced to comply with this Required Action. ---------- Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP

3. Technical Specification 3.2.2 ACTIONS are revised as follows:

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 ~~Be in MODE 2~~Reduce THERMAL POWER to < 25% RTP.	6 hours

4. Technical Specification 3.2.2 SURVEILLANCE REQUIREMENTS are revised as follows:

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	---------- **- NOTE -** Not required to be performed until ~~31 days~~24 hours after ~~the last verification of~~ OPDMS not monitoring parameters. ---------- Verify $F^N_{\Delta H}$ within limits specified in the COLR.	31 effective full power days (EFPD)

COL Appendix A, Technical Specification 3.2.5, On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters, is revised as follows:

1. Technical Specification LCO 3.2.5.a is revised as follows:

LCO 3.2.5 The following parameters shall not exceed their operating limits as specified in the COLR:

a. Peak Linear ~~Power Density~~Heat Rate;

2. Technical Specification 3.2.5 APPLICABILITY is revised as follows:

APPLICABILITY: MODE 1 with THERMAL POWER ~~> 50%~~≥ 25% RTP and with OPDMS monitoring parameters a, b, and c.

3. Technical Specification 3.2.5 ACTIONS are revised as follows:

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to ~~≤ 50%~~< 25% RTP.	4 hours

COL Appendix A, Technical Specification 3.3.1, Reactor Trip System (RTS) Instrumentation, is revised as follows:

1. Technical Specification 3.3.1 SURVEILLANCE REQUIREMENTS are revised as follows:

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	---------- **- NOTES -** ~~1. Adjust nuclear instrument channel in the Protection and Safety Monitoring System (PMS) if absolute difference is > 1% RTP.~~ ~~2~~1. Required to be met within 12 hours after reaching 15% RTP. ~~3~~2. If the calorimetric heat balance is ~~< 70%~~≥ 15% RTP, and if the nuclear instrumentation channel indicated power is: a. lower than the calorimetric measurement by > ~~1%~~5% RTP, then adjust the nuclear instrumentation channel upward to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required. ----------	
	Compare results of calorimetric heat balance to nuclear instrument channel output.	24 hours

2. Technical Specification 3.3.1 SURVEILLANCE REQUIREMENTS are revised as follows:

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	---------- **- NOTES -** 1. Adjust the conversion factor, ΔT°, in the ΔT power calculation ($q_{\Delta T}$) if absolute difference between $q_{\Delta T}$ and the calorimetric measurement is > ~~1%~~3% RTP. 2. Required to be met within 12 hours after reaching 50% RTP. 3. If the calorimetric heat balance is < 70% RTP, and if $q_{\Delta T}$ is: a. lower than the calorimetric measurement by > 5%, then adjust ΔT° to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required. ---------- Compare results of calorimetric heat balance to the ΔT power calculation ($q_{\Delta T}$) output.	24 hours

3. Technical Specification 3.3.1 SURVEILLANCE REQUIREMENTS are revised as follows:

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	---------- **- NOTES -** 1. Adjust nuclear instrument channel in PMS if absolute difference is ≥ ~~3%~~1.5% AFD. 2. Required to be met within 24 hours after reaching 20% RTP. ---------- Compare results of the incore detector measurements to nuclear instrument channel AXIAL FLUX DIFFERENCE.	31 effective full power days (EFPD)

COL Appendix A, Technical Specification 3.7.1, Main Steam Safety Valves (MSSVs)

1. Technical Specification 3.7.1 ACTIONS are revised as follows:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both steam generators with one or more MSSVs inoperable for opening.	A.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours
	AND	
	A.2 ---------- **- NOTE -** Only required in MODE 1. ---------- Reduce the ~~Power Range Neutron Flux – High~~ Overpower ΔT reactor trip setpoints to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours

COL Appendix A, Technical Specification 5.6.3, CORE OPERATING LIMITS REPORT (COLR), is revised as follows:

1. Technical Specification 5.6.3 are revised as follows:

* * *

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

* * *

7a. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004 (Westinghouse Proprietary) and WCAP-16045-NP-A, (Non-Proprietary).

(Methodology for Specification 3.1.3 - Moderator Temperature Coefficient (MTC).)

7b. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007 (Westinghouse Proprietary) and WCAP-16045-NP-A, Addendum 1-A, (Non-Proprietary).

(Methodology for Specification 3.1.3 - Moderator Temperature Coefficient (MTC).)

7c. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997 (Westinghouse Proprietary) .

(Methodology for Specification 3.1.3 - Moderator Temperature Coefficient (MTC).)

UFSAR Tier 2 Subsection 7.2.1.1.2, Nuclear Overpower Trips, is revised as follows:

7.2.1.1.2 Nuclear Overpower Trips

Power Range High Neutron Flux Trip (High Setpoint)

* * * It provides a backup trip to the overtemperature ΔT, overpower ΔT, and power range high positive flux rate trips for protection against excessive core power generation during normal operation and is always active. * * *

Power Range High Positive Flux Rate Reactor Trip

This trip protects the reactor ~~when a sudden abnormal increase in power occurs in two out of the four power range channels. It provides protection against ejection accidents of low worth rods from midpower. It~~against a rapid increase in core power generation during normal operation and is always active. ~~A channel is tripped when rate-sensitive circuits in the channel detect rates of change in nuclear power above the setpoint value.~~ * * * The reactor is tripped when nuclear flux increases rapidly in excess of a predetermined setpoint in two out of the four ~~rate~~power range channels~~ have tripped~~.

UFSAR Tier 2 Subsection 7.2.1.1.3, Core Heat Removal Trips, is revised as follows:

7.2.1.1.3 Core Heat Removal Trips

* * *

Overpower ΔT Trip

The Overpower ΔT reactor trip provides confidence of fuel integrity during overpower conditions, and limits the required range for overtemperature ΔT protection~~, and provides a backup to the power range high neutron flux trip~~.

* * *

UFSAR Tier 2 Subsection 15.0.6, Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses, is revised as follows:

15.0.6 Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses

* * * Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4a. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure 15.0.3-1. As described in Section 7.2 and in Reference 16, the overpower ΔT trip protects the core from exceeding the design overpower limit, and the overtemperature ΔT trip protects the core from exceeding the DNB design limit. As shown on the figure, the overtemperature ΔT setpoint plus the error allowances tracks the core DNB design limits, except that the setpoint includes an upper limit on allowable inlet temperature.

* * *

UFSAR Tier 2 Subsection 15.0.7, Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux, is revised as follows:

15.0.7 Instrumentation Drift and Calorimetric Errors~~, Power Range Neutron Flux~~

~~Examples of the instrumentation uncertainties and calorimetric uncertainties used in establishing the power range high neutron flux setpoint are presented in Table 15.0-5.~~

The calorimetric uncertainty is the uncertainty assumed in the determination of core thermal power as obtained from secondary plant measurements. ~~The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on~~On a daily basis, these secondary plant measurements are compared with the ΔT power signal (Reference 16) and with the total ion chamber current (sum of the top and bottom currents) and those signals are adjusted if necessary for acceptable conformance with the calorimetric power measurement.

* * *

UFSAR Tier 2 Subsection 15.0.11.2, LOFTRAN Computer Code, is revised as follows:

15.0.11.2 LOFTRAN Computer Code

* * * The protection and safety monitoring system is simulated to include reactor trips on high neutron flux, power range high positive flux rate, overtemperature ΔT, high and low pressure, low flow, and high pressurizer level. * * *

* * *

UFSAR Tier 2 Subsection 15.0.16, References, revised as follows:

15.0.16 References

* * *

16. Burnett, Toby, "Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips," APP-GW-GLR-137, Revision 1, February 2011.

* * *

UFSAR Tier 2 Table 15.0-4a (Sheet 1 of 2), Protection and Safety Monitoring System Setpoints and Time Delay Assumed in Accident Analyses, is revised as follows:

Table 15.0-4a (Sheet 1 of 2)
Protection and Safety Monitoring System
Setpoints and Time Delay Assumed in Accident Analyses

Function	Limiting Setpoint Assumed in Analyses	Time Delays (seconds)
Reactor trip on power range high ~~neutron~~ positive flux~~, high setting~~ rate	~~118%~~ 15% with 60-second time constant	0.9

* * *

UFSAR Tier 2 Table 15.0-5, Determination of Maximum Power Range Neutron Flux Channel Trip Setpoint, Based on Nominal Setpoint and Inherent Typical Instrumentation Uncertainties, is revised as follows:

Table 15.0-5 Not Used
~~Determination of Maximum Power Range Neutron Flux Channel Trip Setpoint, Based on Nominal Setpoint and Inherent Typical Instrumentation Uncertainties~~

~~Nominal setpoint (% of rated power)~~		~~109~~
~~**Calorimetric errors in the measurement of secondary system thermal power:**~~		
~~**Variable**~~	~~**Accuracy of Measurement of Variable**~~	~~**Effect on Thermal Power Determination (% of Rated Power)**~~
~~Feedwater temperature~~	~~±3°F~~	
~~Steam pressure (small correction on enthalpy)~~	~~±6 psi~~	
~~Feedwater flow~~	~~±0.5% ΔP instrument span (two channels per steam generator)~~	
~~Assumed calorimetric error~~		~~1.0~~
~~Radial power distribution effects on total ion chamber current~~		~~7.8 (b)*~~
~~Allowed mismatch between power range neutron flux channel and calorimetric measurement~~		~~2.0 (c)*~~
~~Instrumentation channel drift and setpoint reproducibility~~	~~0.4% of instrument span (120% power span)~~	~~0.84(d)*~~
~~Instrumentation channel temperature effects~~		~~0.48(e)*~~
~~*Total assumed error in setpoint (% of rated power): $[(a)^2 + (b)^2 + (c)^2 + (d)^2 + (e)^2]^{1/2}$~~		~~±8.4~~
~~Maximum power range neutron flux trip setpoint assuming a statistical combination of individual uncertainties (% of rated power)~~		~~118~~

UFSAR Tier 2 Table 15.0-6 (Sheet 1 of 5), Plant Systems and Equipment Available for Transient and Accident Conditions, is revised as follows:

Table 15.0-6 (Sheet 1 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.1			
Increase in heat removal from the primary system			
Feedwater system malfunctions that result in an increase in feedwater flow	High-2 Steam Generator Level, Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, manual	High-2 steam generator level produced feedwater isolation and turbine trip	Feedwater isolation valves
Excessive increase in secondary steam flow	Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, manual		
Inadvertent opening of a steam generator safety valve	Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, “S”, manual	Low pressurizer pressure, low compensated steam line pressure, low T_{cold}, low-2 pressurizer level	Core makeup tank, feedwater isolation valves, main steam isolation valves (MSIVs), startup feedwater isolation, accumulators
Steam system piping failure	Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, “S”, manual	Low pressurizer pressure, low compensated steam line pressure, high-2 containment pressure, low T_{cold}, manual	Core makeup tank, feedwater isolation valves, main steam line isolation valves (MSIVs), accumulators, startup feedwater isolation

* * *

UFSAR Tier 2 Table 15.0-6 (Sheet 3 of 5), Plant Systems and Equipment Available for Transient and Accident Conditions, is revised as follows:

Table 15.0-6 (Sheet 3 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
* * *			
Section 15.4			
Reactivity and power distribution anomalies			
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Source range high neutron flux, intermediate range high neutron flux, power range high neutron flux (low setting), power range high neutron flux (high setting), power range high ~~nuclear~~ positive flux rate, manual	-	-
Uncontrolled RCCA bank withdrawal at power	Power range high neutron flux, ~~high~~ power range high positive ~~neutron~~ flux rate, overtemperature ΔT, overpower ΔT, high pressurizer pressure, high pressurizer water level, manual	-	Pressurizer safety valves, steam generator safety valves
* * *			
Startup of an inactive reactor coolant pump at an incorrect temperature	Power range high neutron flux, power range high positive flux rate, low flow (P-10 interlock), manual	-	-

UFSAR Tier 2 Table 15.0-6 (Sheet 4 of 5), Plant Systems and Equipment Available for Transient and Accident Conditions, is revised as follows:

Table 15.0-6 (Sheet 4 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.4 (Cont'd)			
Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high neutron flux, power range high neutron flux, power range high positive flux rate, overtemperature ΔT, manual	Source range flux doubling	CVS to RCS isolation valves, makeup pump suction isolation valves, from the demineralized water transfer and storage system
Spectrum of RCCA ejection accidents	Power range high neutron flux, power range high positive flux rate, manual	-	Pressurizer safety valves

* * *

UFSAR Tier 2 Subsection 15.1.2.2.2, Results, is revised as follows:

15.1.2.2.2 Results

In the case of an accidental full opening of both feedwater control valves with the reactor at zero power and the preceding assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Subsection 15.4.1 for an uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition. Therefore, the results of the analysis are not presented here. If the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25-percent nominal full power, or by the power range high positive flux rate trip.

* * *

Because the power level rises by a maximum of about 8 percent above nominal during the excessive feedwater flow incident, the fuel temperature also rises until after reactor trip occurs. The core heat flux lags behind the neutron flux response because of the fuel rod thermal time constant. Therefore, the peak value does not exceed 118 percent of its nominal value (the assumed ~~high neutron flux~~ overpower ΔT trip setpoint). The peak fuel temperature thus remains well below the fuel melting temperature.

* * *

UFSAR Tier 2 Subsection 15.1.3.1, Identification of Causes and Accident Description, is revised as follows:

15.1.3.1 Identification of Causes and Accident Description

* * *

Protection against an excessive load increase accident is provided by the following protection and safety monitoring system signals:

- Overpower ΔT
- Overtemperature ΔT
- Power range high positive flux rate
- Power range high neutron flux

* * *

UFSAR Tier 2 Subsection 15.1.4.1, Identification of Causes and Accident Description, is revised as follows:

15.1.4.1 Identification of Causes and Accident Description

* * *

The following systems provide the necessary protection against an accidental depressurization of the main steam system (see Subsection 7.2.1.1.2):

* * *

- The overpower reactor trips (power range high positive flux rate, neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the “S” signal

* * *

UFSAR Tier 2 Subsection 15.4.2.1, Identification of Causes and Accident Description, is revised as follows:

15.4.2.1 Identification of Causes and Accident Description

An uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Because the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in a violation of the DNB design basis or excessive linear power. Therefore, to avert damage to the fuel cladding, the protection and safety monitoring system (PMS) is designed to terminate any such transient before the DNBR falls below the design limit (see Section 4.4) or the overpower limit is exceeded.

* * *

The automatic features of the PMS that prevent core damage following the postulated accident include the following:

- Power range neutron flux instrumentation actuates a reactor trip if two out of four divisions exceed ~~an overpower~~ setpoint. In particular, the power range neutron flux instrumentation provides the following reactor trip functions:

  1. Reactor trip on ~~high~~ power range ~~neutron~~high positive flux ~~(high setpoint)~~rate

  2. Reactor trip on ~~high~~ power range ~~positive~~high neutron flux ~~rate~~(high setpoint)

  The ~~latter~~first trip protects the core when a sudden abnormal increase in power is detected in the power range neutron flux channel in two out of four PMS divisions. It provides protection against rapid reactivity insertion ~~rate~~ accidents ~~at mid and low power,~~ and ~~it~~ is always active. The second trip serves as a backup to the overpower ΔT reactor trip and is always active.

- Reactor trip is actuated if any two out of four ΔT power divisions exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressurizer pressure to protect against violating the DNB design basis.
  * * *

* * *

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips:

- ~~High neutron~~Power range high positive flux rate (fixed setpoint)
- High pressurizer pressure (fixed setpoint)
- Low pressurizer pressure (fixed setpoint)
- Overpower and overtemperature ΔT (variable setpoints)

* * *

UFSAR Tier 2 Subsection 15.4.2.2.1, Method of Analysis, is revised as follows:

15.4.2.2.1 Method of Analysis

* * *

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. In performing a conservative analysis for an uncontrolled RCCA bank withdrawal at-power accident, the following assumptions are made:

* * *

- ~~The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power.~~ The power range high positive flux rate trip is assumed to be actuated when the power range neutron flux changes at a rate higher than ~~9%~~15% per second with a ~~two~~60 second rate-lag time constant. The overpower ΔT and overtemperature ΔT trips include~~s~~ adverse instrumentation and setpoint uncertainties. The delays for trip actuation assumed are given in Table 15.0-4a.

* * *

~~The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to the DNBR limit.~~ If RCCA movement causes an adverse effect on the axial core power distribution, then the overtemperature ΔT trip setpoint is decreased as necessary to maintain margin to the DNBR design limit, and the overpower ΔT trip setpoint is decreased as necessary to maintain margin to the overpower limit.

* * *

UFSAR Tier 2 Subsection 15.4.2.2.2, Results, is revised as follows:

15.4.2.2.2 Results

Three reactor trip functions were credited in the analyses to provide protection over the entire range of reactivity insertion rates. These are the power range high positive flux rate, overtemperature ΔT, and overpower ΔT trips.

Figures 15.4.2-1 through 15.4.2-6 show the transient response for a representative rapid (80 pcm/s) RCCA withdrawal incident starting from full power. Reactor trip on power range high ~~neutron~~positive flux rate occurs shortly after the start of the transient. Because this is rapid with respect to the thermal time constants of the fuel, small changes in temperature and pressure result, and the DNB design basis described in Section 4.4 is met.

The transient response for a representative intermediate (34 pcm/s) RCCA withdrawal from full power is shown in Figures 15.4.2-7 through 15.4.2-12. Reactor trip on overpower ΔT occurs preventing the peak heat flux from exceeding 118%. The DNB design basis described in Section 4.4 is met.

The transient response for a representative slow (5 pcm/s) RCCA withdrawal from full power is shown in Figures ~~15.4.2-7~~15.4.2-13 through ~~15.4.2-12~~15.4.2-18. Reactor trip on overtemperature ΔT occurs after a longer period. The rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. The DNB design basis described in Section 4.4 is met.

Figure ~~15.4.2-13~~15.4.2-19 shows the minimum DNBR as a function of reactivity insertion rate from initial full-power operation for minimum and maximum reactivity feedback. Minimum DNBR~~,~~ occurs immediately after rod motion. Three reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the ~~high neutron flux,~~ power range high positive flux rate, overpower ΔT, and overtemperature ΔT ~~channels~~trip functions. The minimum DNBR is greater than the design limit value described in Section 4.4. ~~Note that the high positive flux rate trip was needed for only one case (100% power, minimum reactivity feedback, 110 pcm/s) to prevent the peak heat flux from exceeding 118%.~~

Figures ~~15.4.2-14~~15.4.2-20 and ~~15.4.2-15~~15.4.2-21 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents for minimum and maximum reactivity feedback, starting at 60-percent and 10-percent power, respectively. Minimum DNBR, occurs immediately after rod motion. The results are similar to the 100-percent power case, except ~~as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased and~~ the transient is ~~always~~ terminated by either the power range high positive flux rate or overtemperature ΔT reactor trip ~~for the maximum feedback cases~~. ~~In all cases the~~The minimum DNBR is greater than the design limit value described in Section 4.4.

* * *

Referring to Figure ~~15.4.2-14, for example~~15.4.2-19, it is noted that~~:~~ for transients initiated from full power, three reactor trip functions provide the DNB and overpower protection over the range of reactivity insertion rates analyzed. The overtemperature ΔT trip provides DNB protection except for rapid power excursions. The overpower ΔT trip provides protection for the slow to

moderate power excursions. The power range high positive flux rate trip prevents both overpower and low DNBR for rapid power excursions.

~~A. For high reactivity insertion rates (between 38 pcm/s and 110 pcm/s), reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases.~~

~~B. For minimum reactivity feedback cases that assume reactivity insertion rates of less than 38 pcm/s, protection is provided by the overtemperature ΔT trip.~~

~~C. Reactor trip is initiated by overtemperature ΔT for the entire range of reactivity insertion rates for the maximum reactivity feedback cases.~~

~~D. For most of the minimum feedback cases and all of the maximum feedback cases, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which removes additional heat from the reactor coolant system, sharply decreases the rate of increase of reactor coolant system average temperature. This decrease in the rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation. This causes the overtemperature ΔT setpoint to be reached later, with resulting lower minimum DNBRs.~~

~~For transients initiated from full power (see Figure 15.4.2-13), both minimum and maximum reactivity feedback, the minimum DNBR occurs for the lower reactivity insertion rates that trip on overtemperature ΔT (higher reactivity insertion rates trip on high neutron flux).~~

~~At lower reactivity insertion rates the overtemperature ΔT trip predominates and the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) because for these lower reactivity insertion rates, the power increase is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.~~

* * *

Because the RCCA bank withdrawal at-power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For ~~fast reactivity insertion rates~~rapid power excursions, the overpower transient is fast with respect to the fuel rod thermal time constant and the core heat flux lags behind the neutron flux response. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature still remains below the fuel melting temperature.

For slow ~~reactivity insertion rates~~to moderate power excursions, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by either the overpower ΔT or overtemperature ΔT reactor trip before the DNB design basis is violated. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak centerline temperature remains below the fuel melting temperature.

The reactor is tripped during the RCCA bank withdrawal at-power transient such that the ability of the primary coolant to remove heat from the fuel rods is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

* * *

UFSAR Tier 2 Subsection 15.4.2.3, Conclusions, is revised as follows:

15.4.2.3 Conclusions

The ~~power range neutron flux instrumentation~~overpower ΔT, overtemperature ΔT, and power range high positive flux rate trip functions provide adequate protection over the entire range of possible reactivity insertion rates. The DNB design basis, as defined in Section 4.4, is met for all cases. The maximum reactor coolant system pressure remains below 110% of design.

UFSAR Tier 2 Table 15.4-1 (Sheet 1 of 3), Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies, is revised as follows:

Table 15.4-1 (Sheet 1 of 3)
Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time (seconds)

* * *

Uncontrolled RCCA bank withdrawal at power		
1. Case A – Full power with maximum reactivity feedback	Initiation of uncontrolled RCCA withdrawal at a fast reactivity insertion rate (80 pcm/s)	0.0
	Power range high ~~neutron~~positive flux ~~high~~rate trip point reached	~~6.2~~5.2
	Rods begin to fall into core	~~7.1~~6.1
	Minimum DNBR occurs	~~7.4~~6.4
2. Case B – Full power with maximum reactivity feedback	Initiation of uncontrolled RCCA withdrawal at an intermediate reactivity insertion rate (34 pcm/s)	0.0
	Overpower ΔT setpoint reached	18.0
	Rods begin to fall into core	19.9
	Minimum DNBR occurs	20.1
~~2~~3. Case ~~B~~C – Full power with maximum reactivity feedback	Initiation of uncontrolled RCCA withdrawal at a slow reactivity insertion rate (5 pcm/s)	0.0
	Overtemperature ΔT setpoint reached	568.3
	Rods begin to fall into core	570.3
	Minimum DNBR occurs	570.4

UFSAR Tier 2 Figure 15.4.2-1, Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s), is revised to replace the existing figure as follows:

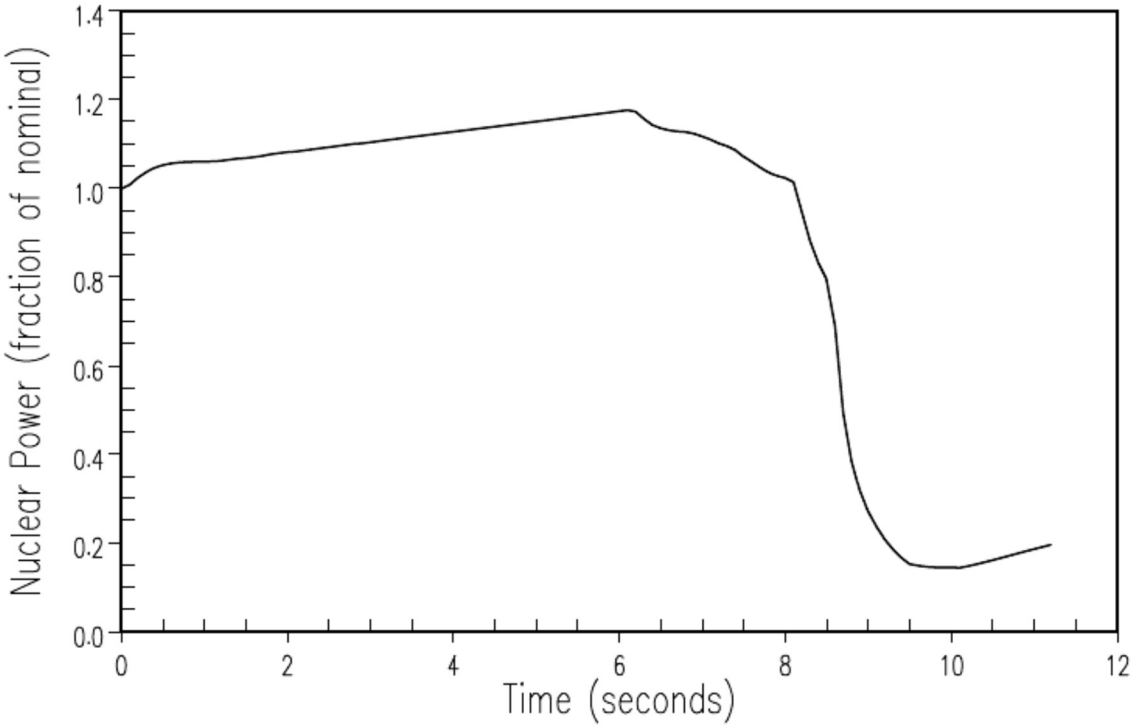

Figure 15.4.2-1
Nuclear Power Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (80 pcm/s)

UFSAR Tier 2 Figure 15.4.2-2, Core Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s), is revised to replace the existing figure as follows:

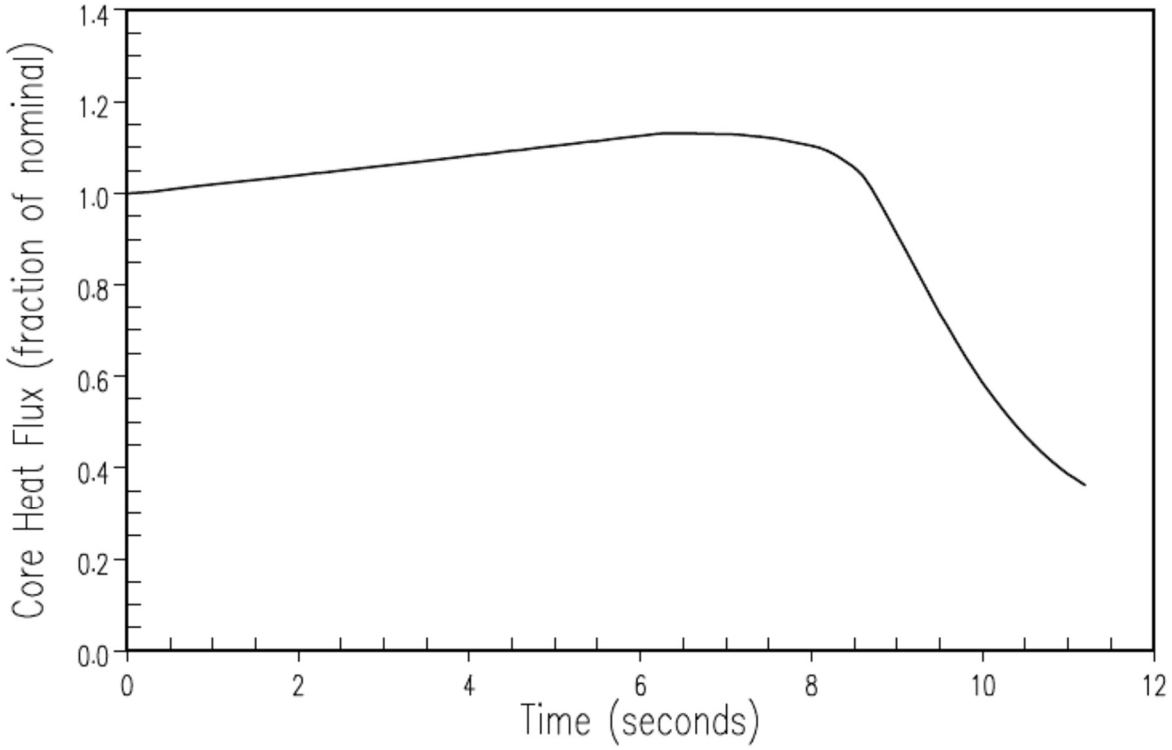

Figure 15.4.2-2
Core Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s)

UFSAR Tier 2 Figure 15.4.2-3, Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s), is revised to replace the existing figure as follows:

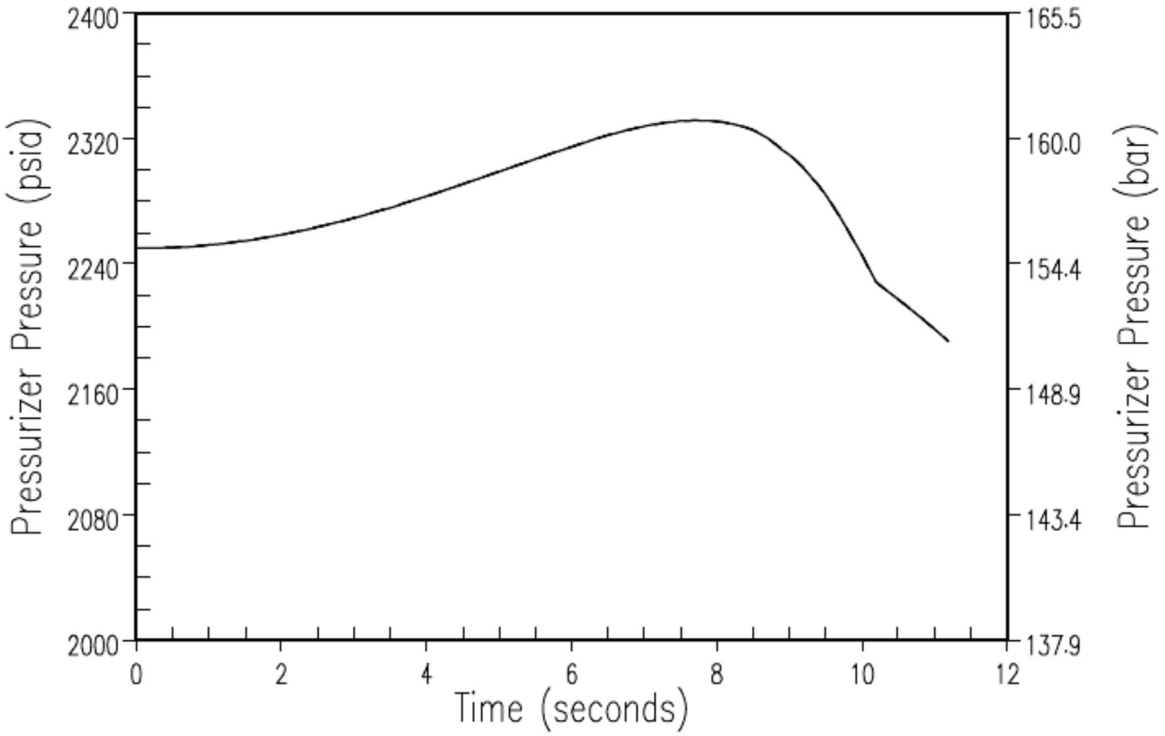

Figure 15.4.2-3
Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s)

UFSAR Tier 2 Figure 15.4.2-4, Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s), is revised to replace the existing figure as follows:

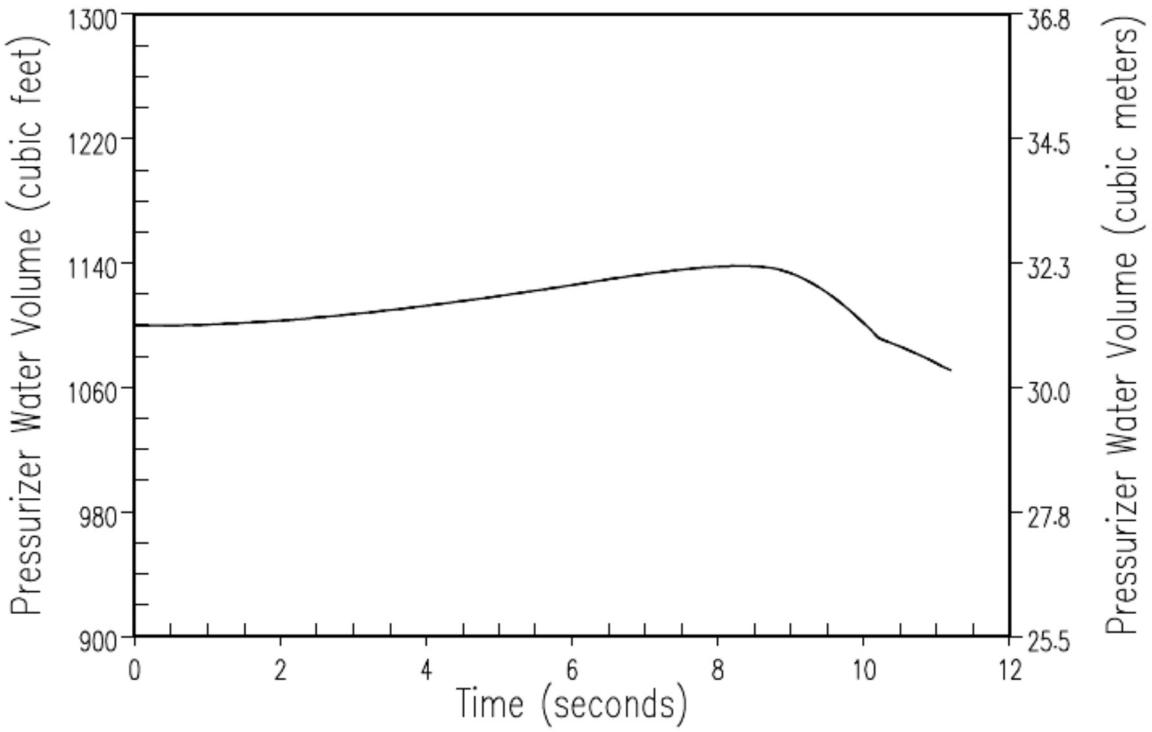

Figure 15.4.2-4
Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s)

UFSAR Tier 2 Figure 15.4.2-5, Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s), is revised to replace the existing figure as follows:

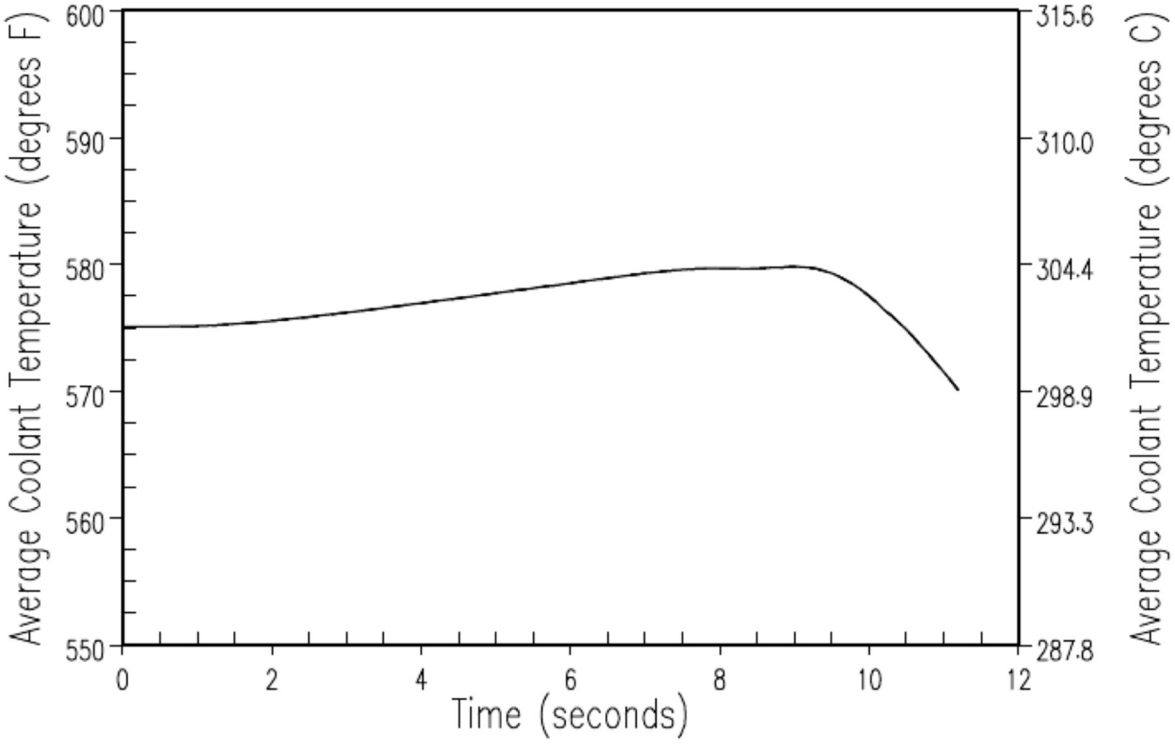

Figure 15.4.2-5
Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s)

UFSAR Tier 2 Figure 15.4.2-6, DNBR Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s), is revised to replace the existing figure as follows:

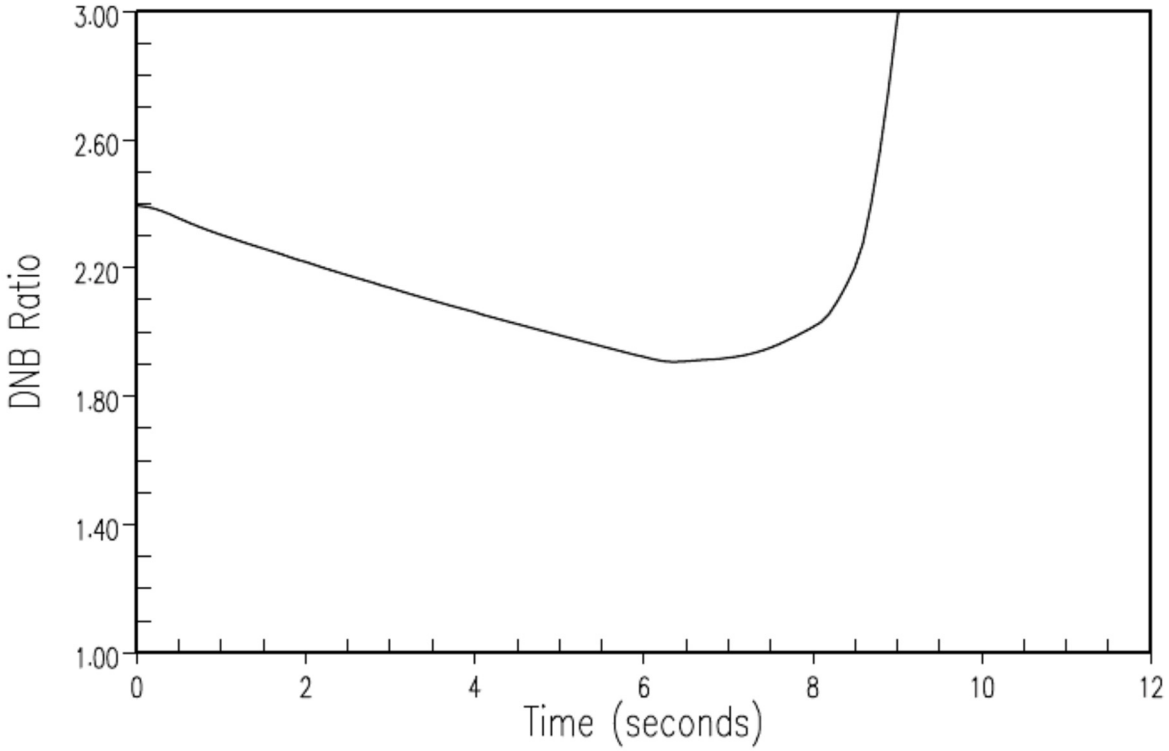

Figure 15.4.2-6
DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (80 pcm/s)

A new UFSAR Tier 2 Figure 15.4.2-7, Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s), is inserted as follows:

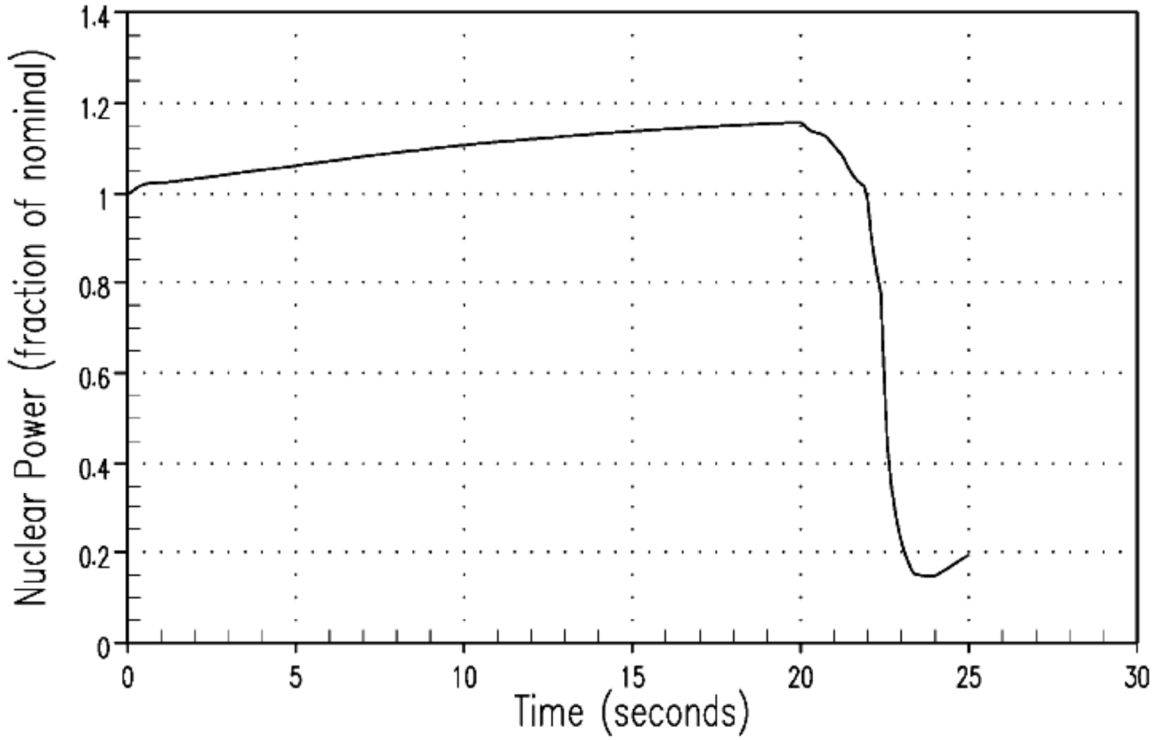

Figure 15.4.2-7
Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

A new UFSAR Tier 2 Figure 15.4.2-8, Core Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s), is inserted as follows:

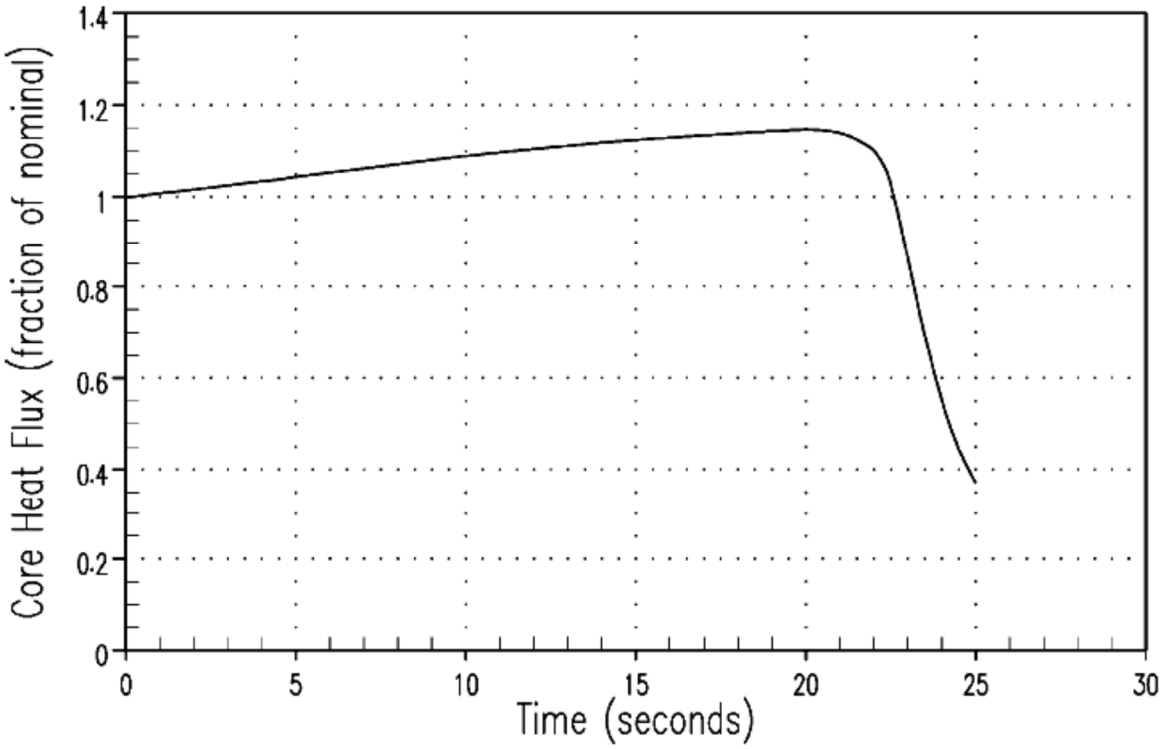

Figure 15.4.2-8
Core Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

A new UFSAR Tier 2 Figure 15.4.2-9, Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s), is inserted as follows:

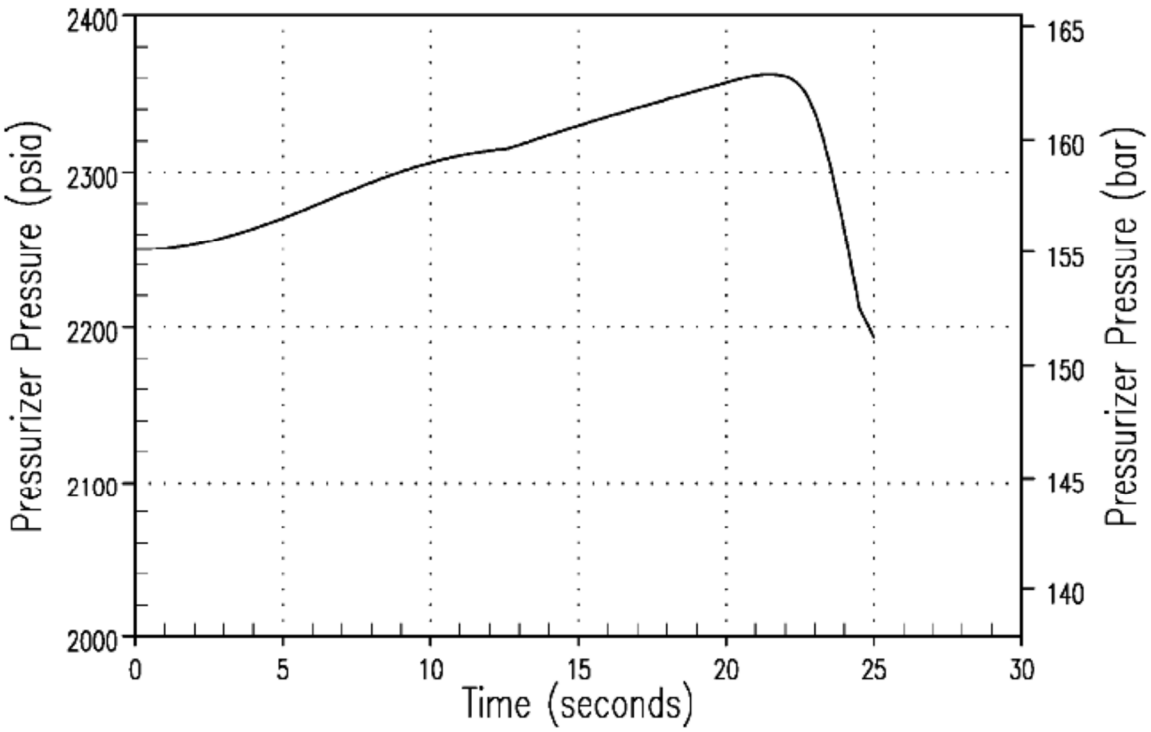

Figure 15.4.2-9
Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

A new UFSAR Tier 2 Figure 15.4.2-10, Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s), is inserted as follows:

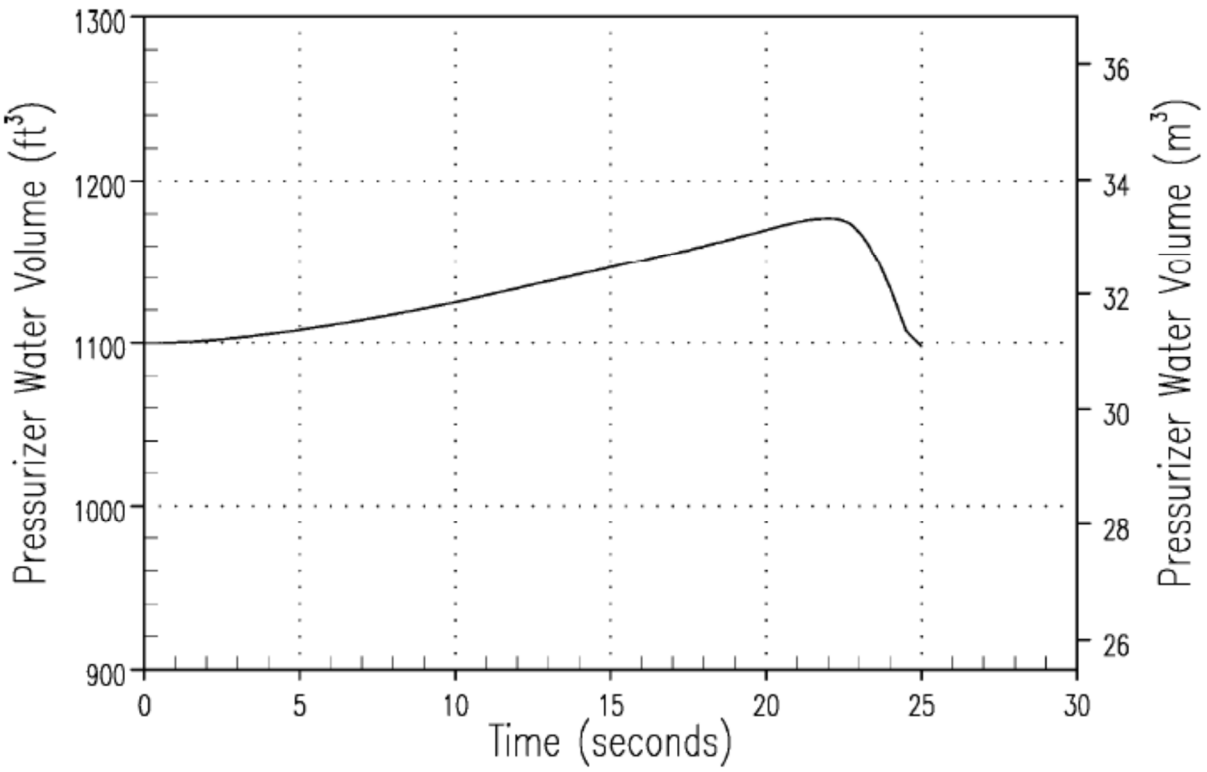

Figure 15.4.2-10
Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

A new UFSAR Tier 2 Figure 15.4.2-11, Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s), is inserted as follows:

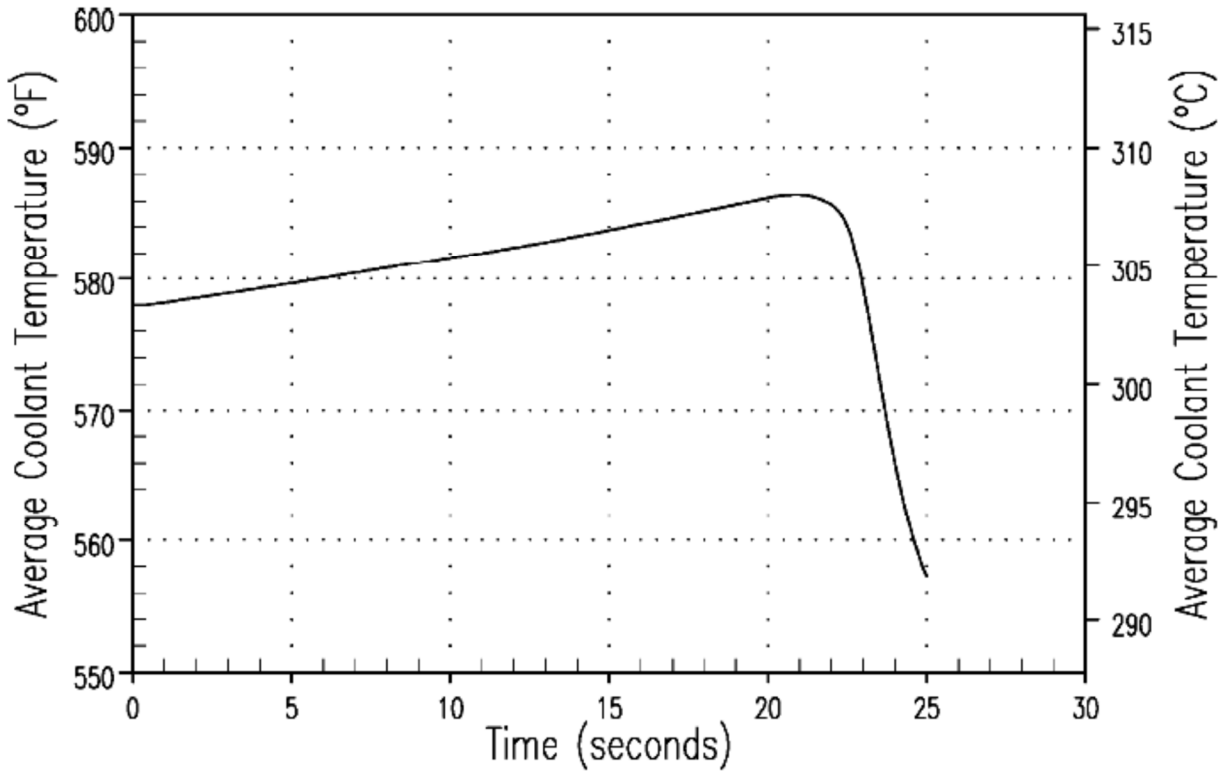

Figure 15.4.2-11
Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

A new UFSAR Tier 2 Figure 15.4.2-12, DNBR Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s), is inserted as follows:

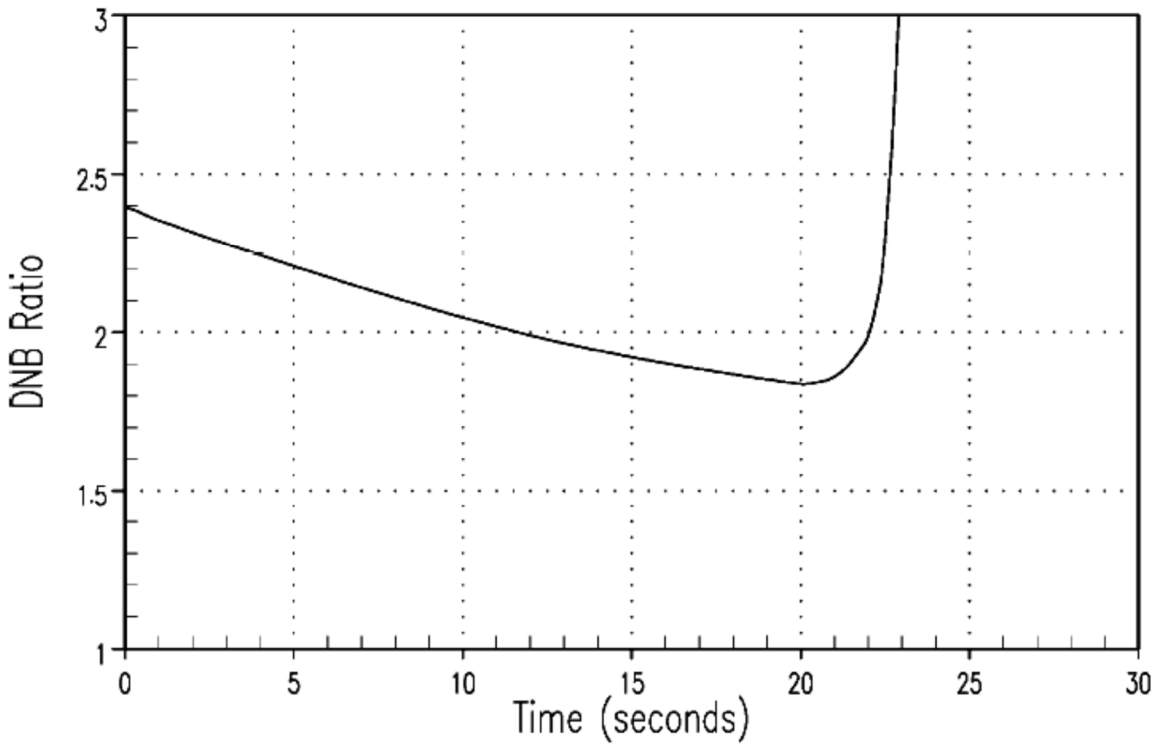

Figure 15.4.2-12
DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (34 pcm/s)

Existing UFSAR Tier 2 Figure 15.4.2-7, Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s), is renumbered and revised to replace the existing figure as follows:

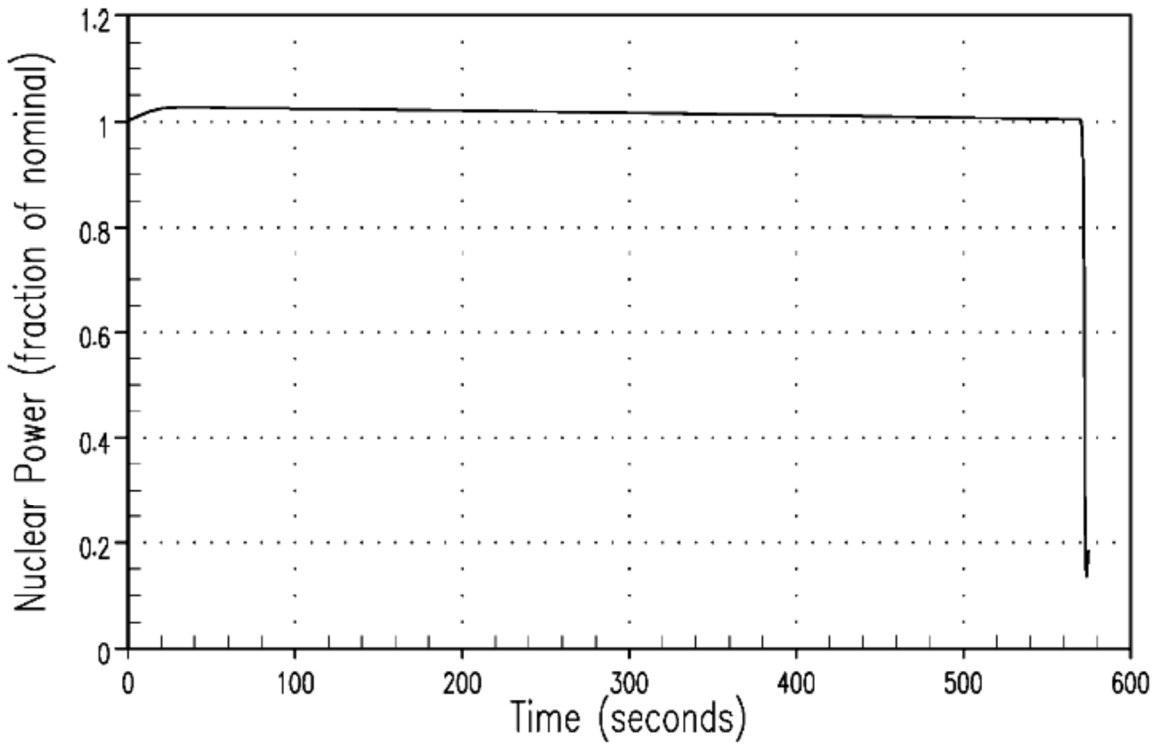

Figure 15.4.2-~~7~~13
Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

Existing UFSAR Tier 2 Figure 15.4.2-8, Core Heat Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s), is renumbered and revised to replace the existing figure as follows:

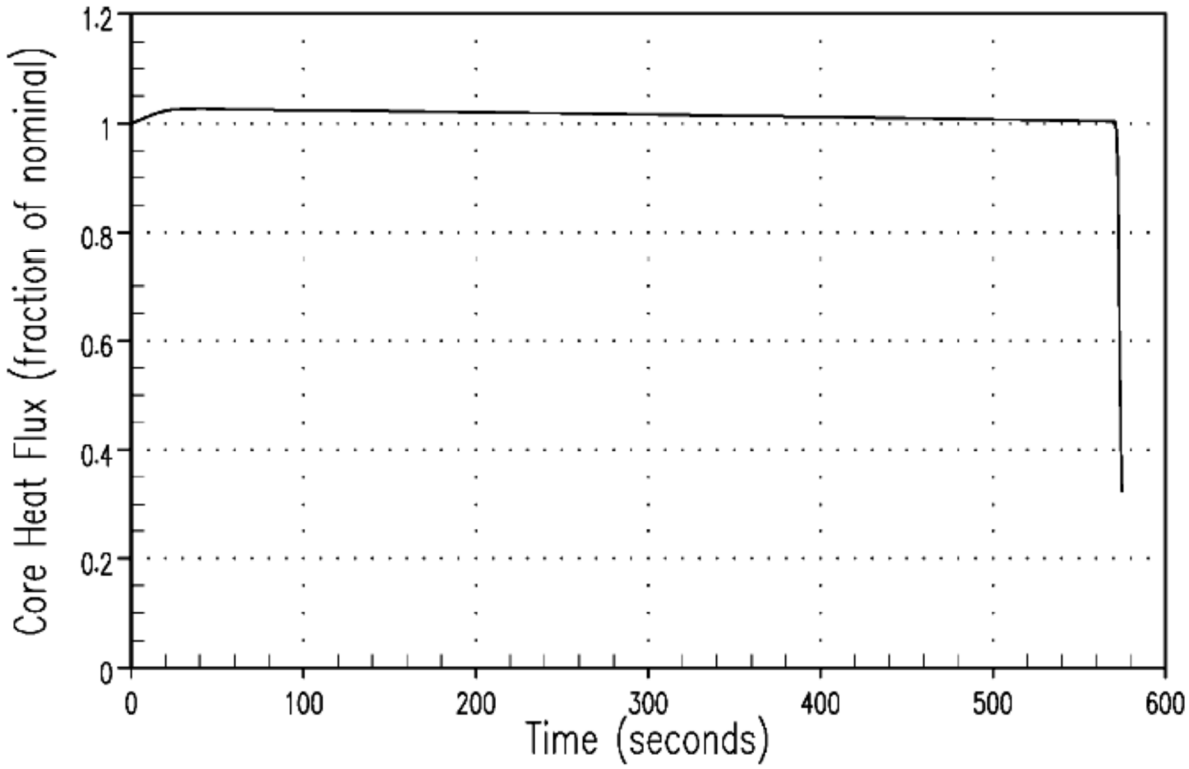

Figure 15.4.2-~~8~~14
Core Heat Flux Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (5 pcm/s)

Existing UFSAR Tier 2 Figure 15.4.2-9, Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s), is renumbered and revised to replace the existing figure as follows:

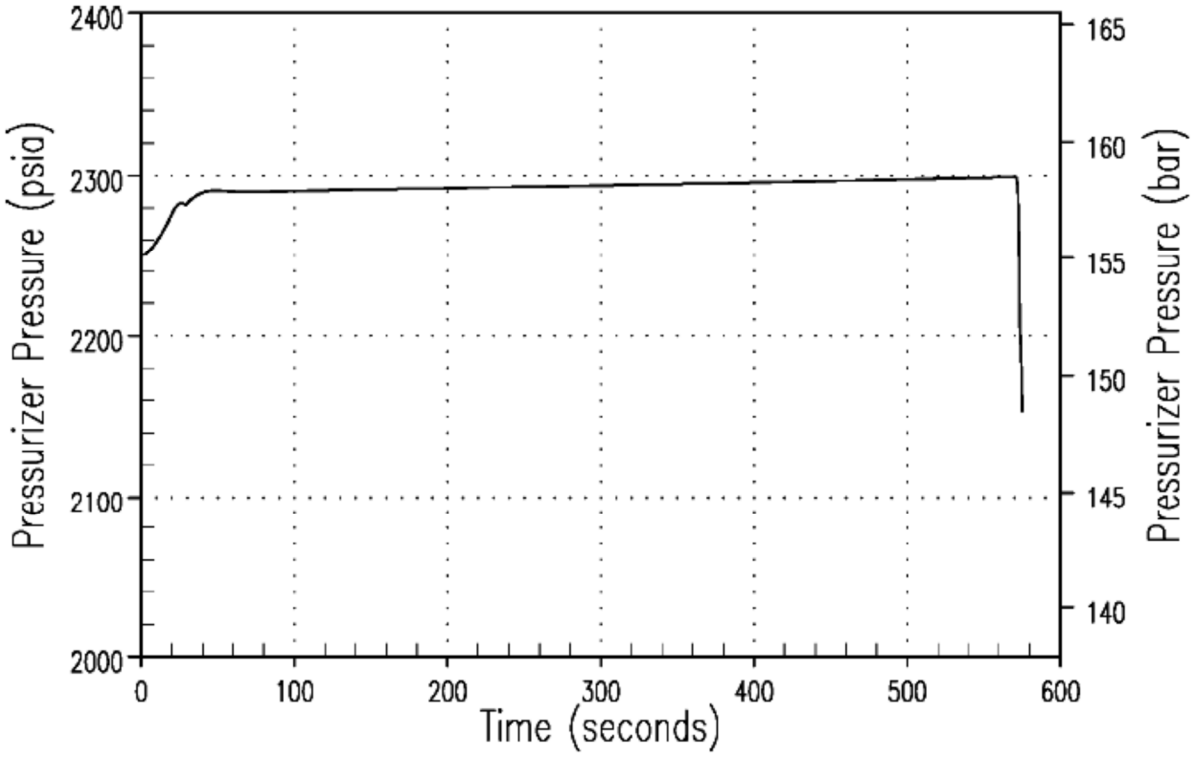

Figure 15.4.2-~~9~~15
Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

Existing UFSAR Tier 2 Figure 15.4.2-10, Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s), is renumbered and revised to replace the existing figure as follows:

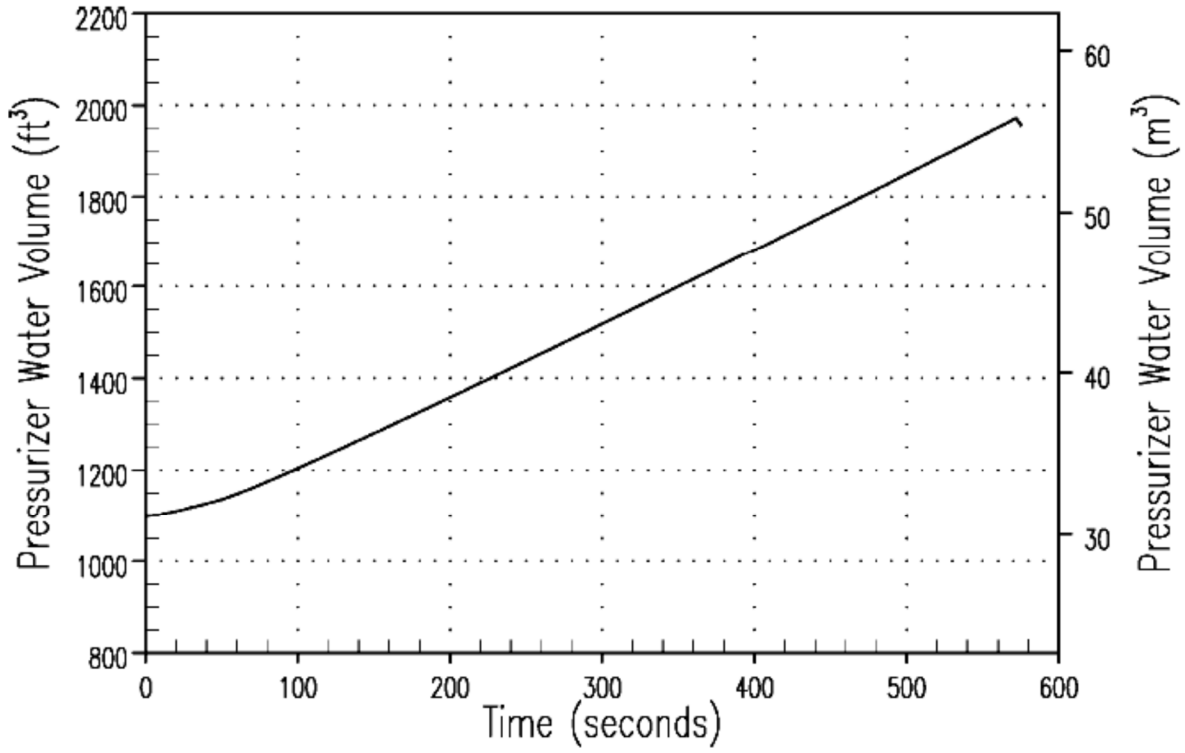

Figure 15.4.2-~~10~~16
Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

Existing UFSAR Tier 2 Figure 15.4.2-11, Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s), is renumbered and revised to replace the existing figure as follows:

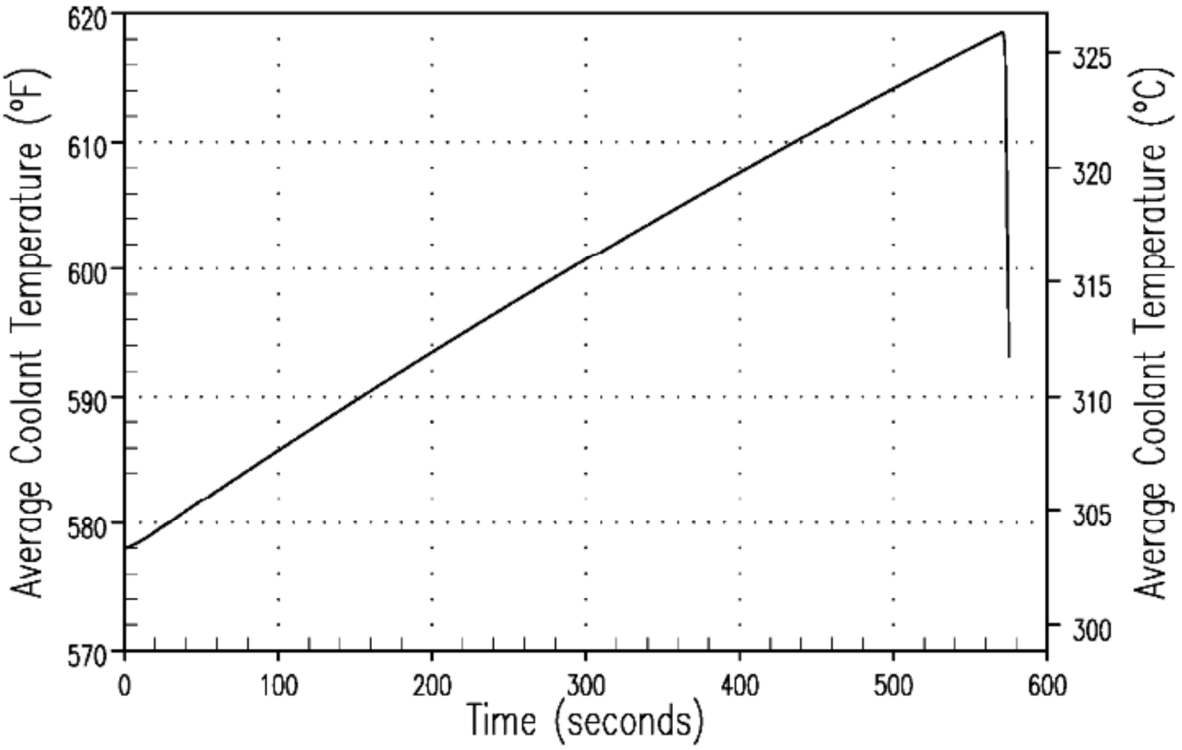

Figure 15.4.2-~~11~~17
Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

Existing UFSAR Tier 2 Figure 15.4.2-12, DNBR Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s), is renumbered and revised to replace the existing figure as follows:

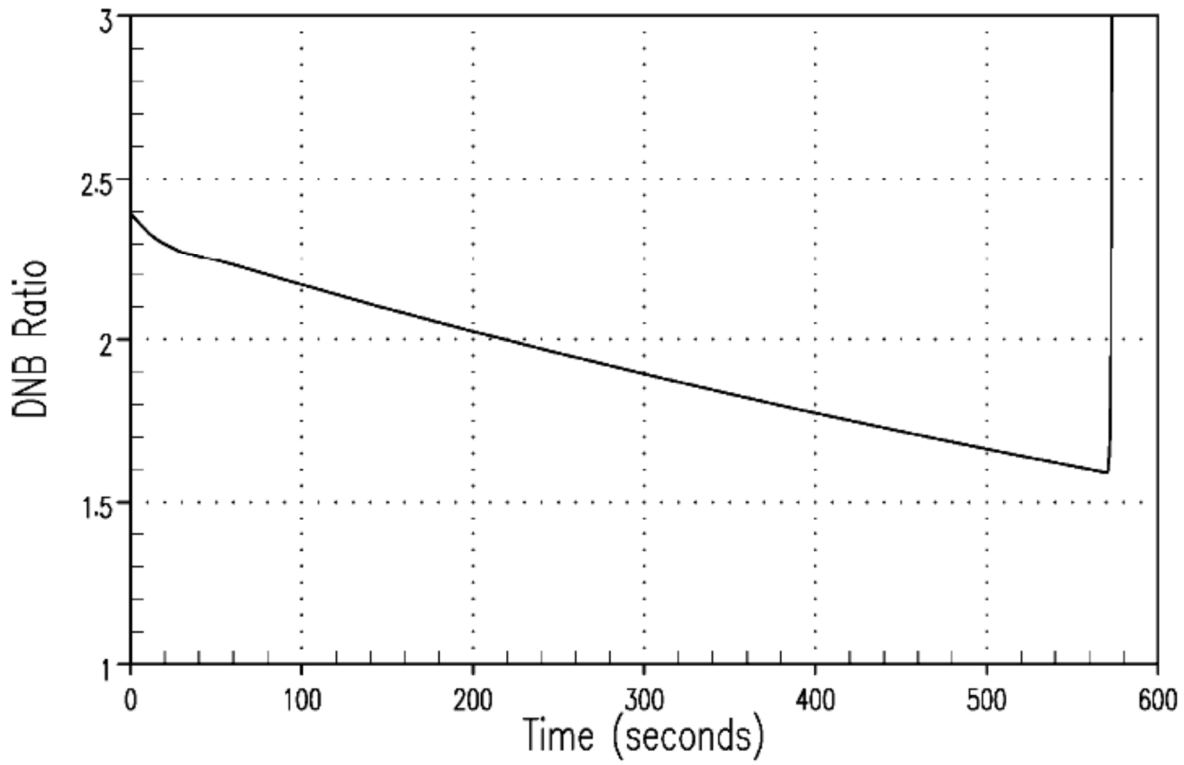

Figure 15.4.2-~~12~~18
DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (5 pcm/s)

Existing UFSAR Tier 2 Figure 15.4.2-13, Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 100-percent Power, is renumbered and revised to replace the existing figure as follows:

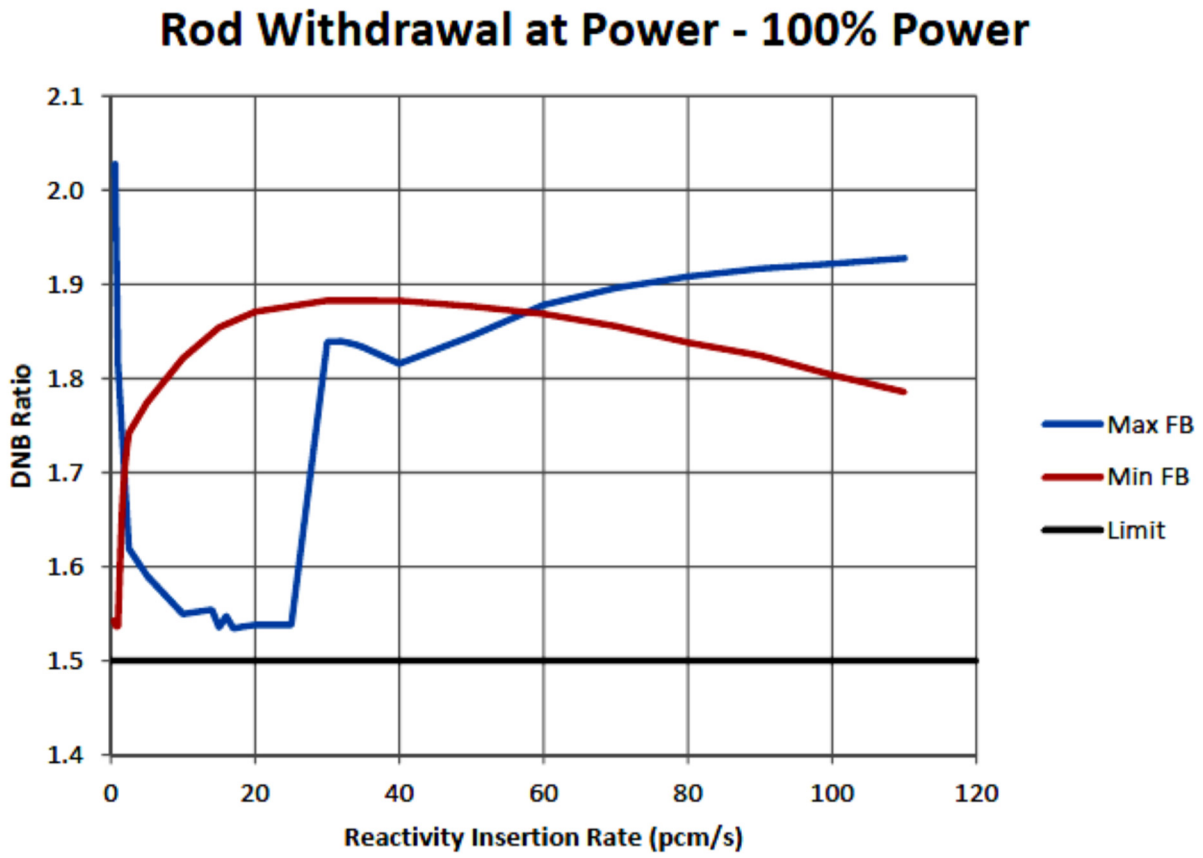

Figure 15.4.2-~~13~~19
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 100-percent Power

Existing UFSAR Tier 2 Figure 15.4.2-14, Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 60-percent Power, is renumbered and revised to replace the existing figure as follows:

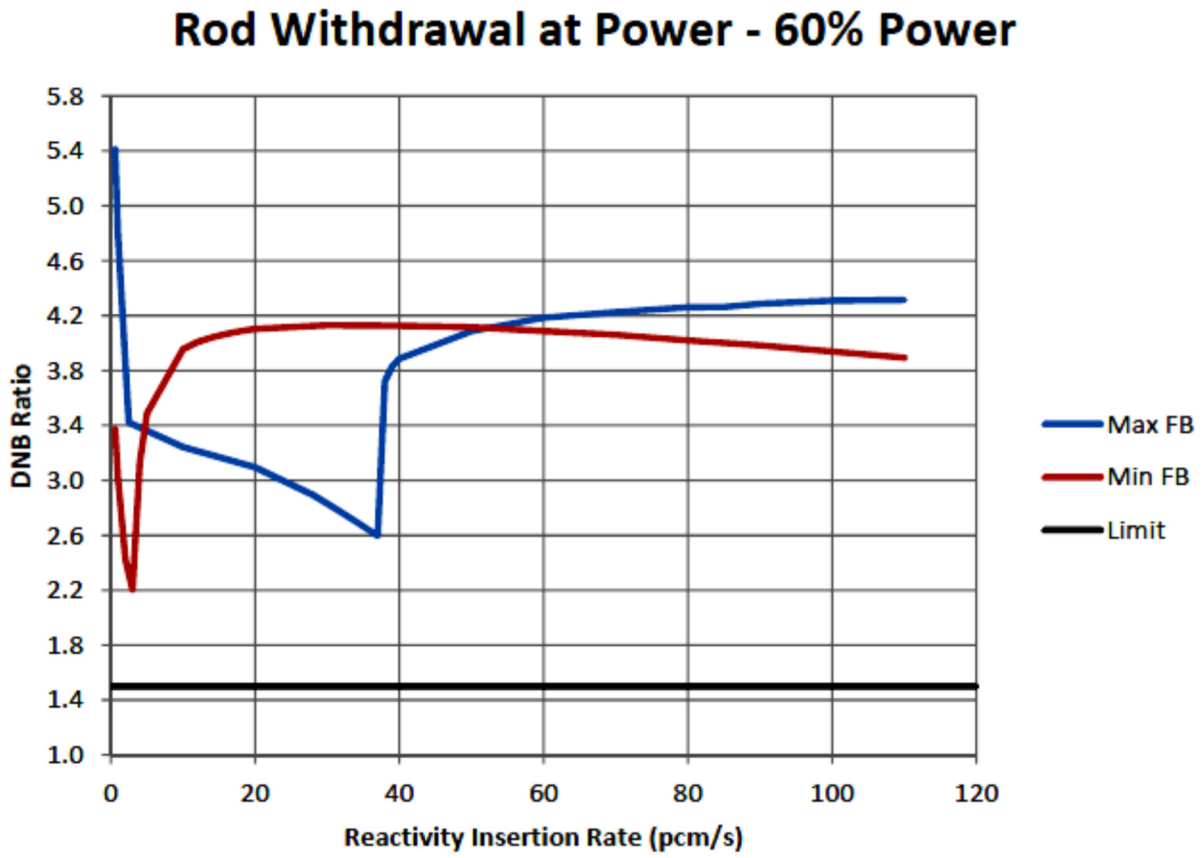

Figure 15.4.2-~~14~~20
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 60-percent Power

Existing UFSAR Tier 2 Figure 15.4.2-15, Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 10-percent Power, is renumbered and revised to replace the existing figure as follows:

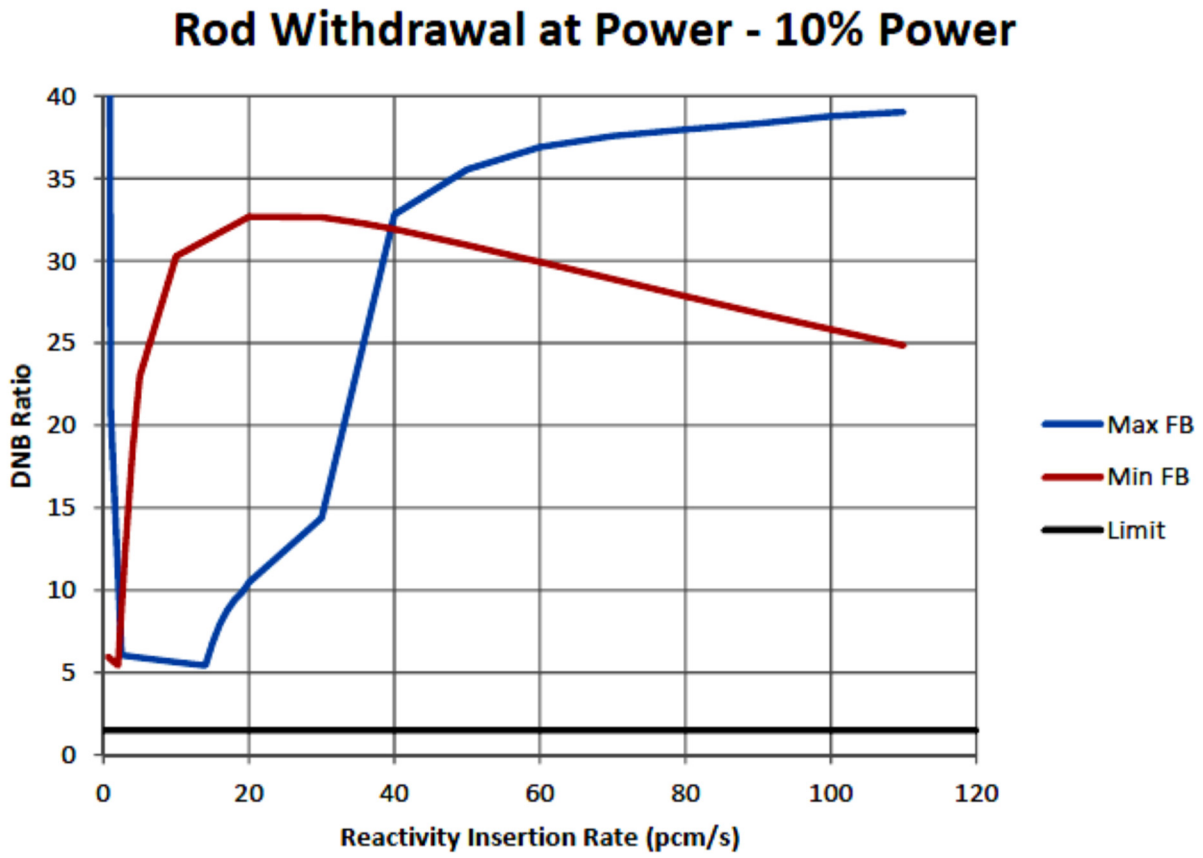

Figure 15.4.2-~~15~~21
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 10-percent Power

Southern Nuclear Operating Company

ND-17-1495

Enclosure 3

Vogtle Electric Generating Plant Units 3 and 4

Conforming Changes to the Technical Specifications Bases

(For Information Only)

(LAR-17-031)

Additions identified by blue underlined text.
Deletions identified by red strikethrough of text.
* * * indicates omitted existing text that is not shown.

(This Enclosure consists of 19 pages, including this cover page)

Technical Specification Bases 3.1.3, Moderator Temperature Coefficient (MTC), are revised as follows:

BASES

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. * * *

* * *

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

During the core safety evaluation, MTC values are predicted at selected burnup accumulation during operations. Based on these predictions, administrative limits may be required on maximum RCS boron concentration and control bank position to maintain the MTC at less than or equal to the upper MTC limit provided in the COLR. These administrative limits consist of tables of the maximum RCS boron concentration versus control rod position, power level, and cycle burnup. A means for adjusting the maximum RCS boron concentrations is also provided to account for differences between the MTC measured during startup physics testing and the corresponding predicted value. The RCS boron concentration needed to maintain a critical core state may increase with cycle burnup accumulation due to burnable absorber depletion. However, as the cycle burnup continues to accumulate, the effect of fuel depletion on core reactivity will exceed the effect of burnable absorber depletion and the RCS boron concentration needed to maintain criticality will decrease. The reduction in critical boron concentration causes the MTC to become more negative. The time in cycle life at which the calculated MTC will meet the upper MTC limit can be determined. At this point in the cycle and beyond, the administrative limits on maximum RCS boron concentration and control bank position are no longer required to meet the upper MTC limit.

The BOC limit and the EOC limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

ACTIONS

A.1

If the upper MTC limit is violated, administrative ~~withdrawal limits for control banks must be established~~ limits are placed on the combination of

maximum RCS boron concentration and control bank position to maintain the MTC within its upper limit~~s~~. ~~The MTC becomes more negative with control bank insertion and decreased boron concentration.~~ A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required ~~bank withdrawal~~ administrative limits.

~~Early in an operating cycle, the RCS boron concentration needed to maintain a critical core state may increase with burnup due to burnable absorber depletion. However, as the cycle burnup continues to increase, the effect of fuel depletion on core reactivity will exceed the effect of burnable absorber depletion and the RCS boron concentration needed to maintain criticality will begin to decrease with burnup. The decreasing critical boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life, Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.~~

B.1

If the required administrative ~~withdrawal~~ limits ~~at BOC~~ are not established within 24 hours, the unit must be placed in MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC which is less negative than that assumed in safety analyses.

* * *

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

* * *

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the MTC limit of the LCO. ~~If required,~~ The measurement results and predicted design values ~~can be~~ are used to ~~establish~~ determine if administrative ~~withdrawal~~ limits ~~for control banks~~ on the combination of maximum RCS boron concentration and control bank position are required at any time in the cycle.

SR 3.1.3.2

In similar fashion, the LCO ~~demands~~ requires that the MTC be less negative than the specified value for EOC full power conditions. * * *

* * *

SR 3.1.3.2 is modified by ~~a~~ two Notes. The first Note states that an exemption from the measurement may be allowed provided applicable criteria provided in the COLR are met. This is based on the requirements of WCAP-13749-P-A (Reference 4). Reference 4 allows that if a specified revised prediction of the MTC and limits for several core parameters measured during the cycle are within specified bounds, the end of life (EOL) MTC measurement need not be made. The second Note states that the Surveillance limit for RTP boron concentration of 60 ppm is conservative. * * *

REFERENCES

* * *

4. WCAP-13749-P-A, "Safety Evaluation Supporting The Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997.

Technical Specification Bases 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology), are revised as follows:

BASES

BACKGROUND

* * *

With the OPDMS monitoring parameters, peak linear ~~power density~~ heat rate (which is proportional to $F_Q(Z)$) is measured continuously. * * *

* * *

APPLICABILITY

When the OPDMS is not monitoring parameters and core power distribution parameters cannot be continuously monitored, it is necessary to determine $F_Q(Z)$ on a periodic basis. Furthermore, the $F_Q(Z)$ limits must be maintained in MODE 1 with THERMAL POWER ≥ 25% RTP to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in MODE 1 with THERMAL POWER < 25% RTP is not required because local power peaking at very low power levels is not limiting with respect to design basis accident analyses and because the incore instrumentation system may not be able to provide an accurate measurement of the core flux distribution at very low THERMAL POWER levels. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

* * *

~~A.2~~

~~A reduction of the Power Range Neutron Flux – High Trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux – High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux – High trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux – High trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux – High trip setpoints.~~

A.~~3~~2

* * * The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.~~3~~2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. * * *

A.~~4~~3

* * *

Condition A is modified by a Note that requires Required Action A.~~4~~3 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.~~4~~3. * * *

* * *

~~B.2~~

~~A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.~~

B.~~3~~2

* * *

B.~~4~~3

* * *

Condition B is modified by a Note that requires Required Action B.~~4~~3 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition A is exited prior to performing Required Action B.~~4~~3. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

C.1

If Required Actions A.1 through A.~~4~~3 or B.1 through B.~~4~~3 are not met within their associated Completion Times, the plant must be placed in a

MODE or condition in which the LCO requirements are not applicable. This is done by ~~placing the plant in at least MODE 2~~ reducing THERMAL POWER to < 25% RTP within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach ~~MODE 2~~ < 25% RTP from full power operation in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

* * *

The SR 3.2.1.3 Note and SR 3.2.1.4 Note 1 apply to the situation where the OPDMS is no longer monitoring parameters while the plant is in MODE 1. Without the continuous monitoring capability of the OPDMS, F_Q limits must be monitored on a periodic basis. The first measurement must be made within ~~31 days~~ 24 hours of the ~~most recent date where the OPDMS data has verified peak linear power density (and therefore also F_Q) to be within its limit~~ OPDMS being declared not monitoring parameters. This is expected to be performed using the most recent contingency surveillance report created automatically during OPDMS monitoring. Subsequent surveillances are required consistent with the 31 ~~day Surveillance~~ EFPD Frequency.

* * *

Technical Specification Bases 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), are revised as follows:

BASES

APPLICABLE SAFETY ANALYSES

* * *

The fuel is protected in part by Technical Specifications, which provide assurance that the initial conditions assumed in the safety and accident analyses remain valid. With the OPDMS monitoring parameters, peak linear ~~power density~~ heat rate and $F^N_{\Delta H}$ are directly monitored. * * *

* * *

APPLICABILITY

When the OPDMS is not monitoring parameters and core power distribution parameters cannot be continuously monitored, it is necessary to monitor $F^N_{\Delta H}(Z)$ on a periodic basis. Furthermore, $F^N_{\Delta H}$ limits must be maintained in MODE 1 with THERMAL POWER ≥ 25% RTP to preclude core power distributions from exceeding the fuel design limits for DNBR and peak cladding temperature (PCT). * * *

ACTIONS

* * *

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the ~~Power Range Neutron Flux - High~~ Overpower ΔT trip setpoints to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. * * *

* * *

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by ~~placing the plant in at least MODE 2~~ reducing THERMAL POWER to < 25% RTP

within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach ~~MODE 2~~ < 25% RTP from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 and SR 3.2.2.2

* * *

SR 3.2.2.2 is modified by a Note, which applies to the situation where the OPDMS is no longer monitoring parameters. Without the continuous monitoring capability of the OPDMS, $F^N_{\Delta H}$ limits must be monitored on a periodic basis. The first measurement must be made within ~~31 days~~ 24 hours of the ~~most recent date where the OPDMS data has verified parameters to be within limits~~ OPDMS being declared not monitoring parameters. This is expected to be performed using the most recent contingency surveillance report created automatically during OPDMS monitoring. Subsequent surveillances are required consistent with the 31 ~~day Surveillance~~ EFPD Frequency.

Technical Specification Bases 3.2.4, Quadrant Power Tilt Ratio (QPTR), are revised as follows:

BASES

BACKGROUND * * *

The power density at any point in the core must be limited so that the fuel design criteria are maintained. With the OPDMS monitoring parameters, the peak linear ~~power density~~ heat rate is continuously and directly monitored. * * *

Technical Specification Bases 3.2.5, On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters, are revised as follows:

BASES

BACKGROUND

* * *

* * * In addition, limiting the peak linear ~~power density~~ heat rate during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the peak cladding temperature (PCT) limit of 2200°F is not exceeded.

The definition of certain quantities used in these specifications are as follows:

Peak linear ~~power density~~ heat rate	Peak linear ~~power density~~ heat rate (axially dependent) as measured in kw/ft.

* * *

APPLICABLE SAFETY ANALYSES

* * *

Limits on linear power density or peak kw/ft assure that the peak linear ~~power density~~ heat rate assumed as a base condition in the LOCA analyses is not exceeded during normal operation.

* * *

LCO

* * *

Peak linear ~~power density~~ heat rate limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. * * *

* * *

APPLICABILITY

The OPDMS-monitored power distribution ~~parameter~~ limits with OPDMS monitoring parameters a, b, and c must be maintained in MODE 1 ~~above 50%~~ with THERMAL POWER ≥ 25% RTP to preclude core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES, and MODE 1 ~~below 50%~~ with THERMAL POWER < 25% RTP, is not required because there is either insufficient stored energy in the fuel or insufficient energy transferred to the reactor

coolant to require a limit on the distribution of core power. The OPDMS monitoring of SDM is applicable in MODES 1 and 2 with $k_{eff} \geq 1.0$.

Specifically for $F^N_{\Delta H}$ and DNBR, the design bases accidents (DBAs) that are sensitive to $F^N_{\Delta H}$ and DNBR in other MODES (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ and DNBR in these modes.

In addition to the alarms discussed in the LCO section above (alarms on OPDMS-monitored power distribution parameters), there is an alarm indicating the potential for the OPDMS ~~itself~~ to be not monitoring parameters.

* * *

ACTIONS

A.1

* * * The 1 hour operator ACTION requirement to restore the parameter to within limits is consistent with the basis for the anticipated operational occurrences and provides time to assess if there are instrumentation problems. It also allows the possibility to restore the parameter to within limits by control rod ~~cluster control assembly (RCCA)~~ motion if this is possible. * * *

B.1

* * *

If the parameters cannot be returned to within limits as power is being reduced, THERMAL POWER must be reduced to < ~~50%~~ 25% RTP where the LCOs are no longer applicable.

The Completion Time of 4 hours provides an acceptable time to reduce power in an orderly manner and without allowing the ~~plant to remain outside~~ reactor to exceed the $F^N_{\Delta H}$ limits for an extended period of time.

* * *

Technical Specification Bases 3.3.1, Reactor Trip System (RTS) Instrumentation, are revised as follows:

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

* * *

* * *

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Power Range Neutron Flux

* * *

a. Power Range Neutron Flux – High

The Power Range Neutron Flux – High trip Function ~~ensures that~~ provides backup protection ~~is provided, from all~~ , at high power levels, against a positive reactivity excursion during power operations. * * *

* * *

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux – High trip must be OPERABLE. This Function ~~will~~ provides a backup trip to Overtemperature ΔT, Overpower ΔT, and Power Range Neutron Flux - High Positive Rate to terminate the reactivity excursion and shutdown the reactor ~~prior to reaching a power level that could damage the fuel~~. * * *

* * *

2. Power Range Neutron Flux – High Positive Rate

The Power Range Neutron Flux – High Positive Rate trip Function ensures that protection is provided against moderate to rapid increases in neutron flux which are characteristic of a an uncontrolled rod cluster control assembly (RCCA) withdrawal or a drive rod housing rupture and the accompanying ejection of the RCCA. This Function ~~compliments the Power Range Neutron Flux – High and Low trip Functions to ensure that the criteria are met for a rod ejection from the power range~~ works with the Overtemperature ΔT and Overpower

ΔT trip functions, through overlap, to prevent overpower and low DNBR for moderate to rapid power excursions. The Power Range Neutron Flux - High Positive Rate trip uses the same channels as discussed for Function ~~2~~1 above.

The LCO requires four Power Range Neutron Flux – High Positive Rate channels to be OPERABLE. In MODE 1 or 2, when there is a potential to add ~~a large amount of~~ positive reactivity from an uncontrolled RCCA withdrawal or a rod ejection accident (REA), the Power Range Neutron Flux – High Positive Rate trip must be OPERABLE. * * *

3. Overtemperature ΔT

The Overtemperature ΔT trip Function ensures that protection is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. This Function works with the Power Range Neutron Flux - High Positive Rate trip to provide DNB protection for low to moderate positive reactivity excursions due to uncontrolled RCCA withdrawal events. The inputs to the Overtemperature ΔT trip include all combinations of pressure, power, coolant temperature, and axial power distribution, assuming full reactor coolant flow. * * *

* * *

4. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip function ~~and provides a backup to the Power Range Neutron Flux – High Setpoint trip~~. This Function works with the Power Range Neutron Flux - High Positive Rate trip to limit overpower for low to moderate positive reactivity excursions due to uncontrolled RCCA withdrawal events. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the same ΔT power signal generated for the Overtemperature ΔT. * * *

* * *

SURVEILLANCE REQUIREMENTS

* * *

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance to the nuclear instrumentation channel output every 24 hours. If the calorimetric measurement ~~between 70% and 100%~~ at ≥ 15% RTP, differs from the nuclear instrument channel output by > ~~1%~~ 5% RTP, the nuclear instrument channel is not declared inoperable, but must be adjusted. If the nuclear instrument channel output cannot be properly adjusted, the channel is declared inoperable.

~~Three~~ Two Notes modify SR 3.3.1.2. The first Note ~~indicates that the nuclear instrument channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the nuclear instrument channel output and the calorimetric measurement between 70% and 100% RTP is > 1% RTP. The second Note~~ clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels the calorimetric data from feedwater flow venturi measurements are less accurate. The ~~third~~ second Note is required because, at power levels ~~between 15% and 70%~~ ≥ 15% RTP, calorimetric uncertainty and control rod insertion create the potential for miscalibration of the nuclear instrumentation channel ~~in cases where the channel is adjusted downward to match the calorimetric power~~. Therefore, if the calorimetric heat measurement is ~~less than 70%~~ ≥ 15% RTP, and if the nuclear instrumentation channel indicated power is lower than the calorimetric measurement by > ~~1%~~ 5% RTP, then the nuclear instrumentation channel shall be adjusted upward to match the calorimetric measurement. No nuclear instrumentation channel adjustment is required if the nuclear instrumentation channel is higher than the calorimetric measurement ~~(see Westinghouse Technical Bulletin NSD-TB-92-14, Rev. 1.)~~

* * *

Together, these factors demonstrate the change in the absolute difference between nuclear instrumentation and heat balance calculated powers rarely exceeds ~~1%~~ 5% RTP in any 24 hours period.

* * *

SR 3.3.1.3

SR 3.3.1.3 compares the calorimetric heat balance to the calculated ΔT power ($q_{\Delta T}$) in each Division every 24 hours. If the calorimetric

measurement between 70% and 100% RTP, differs from the calculated ΔT power by > ~~1%~~ 3% RTP, the Function is not declared inoperable, but the conversion factor, ΔT°, must be adjusted. If ΔT° cannot be properly adjusted, the Function is declared inoperable in the affected Division(s).

Three Notes modify SR 3.3.1.3. The first Note indicates that ΔT° shall be adjusted consistent with the calorimetric results if the absolute difference between the calculated ΔT power and the calorimetric measurement between 70% and 100% RTP is > ~~1%~~ 3% RTP.

* * *

SR 3.3.1.4

SR 3.3.1.4 compares the AXIAL FLUX DIFFERENCE determined using the incore system to the nuclear instrument channel AXIAL FLUX DIFFERENCE every 31 effective full power days (EFPD) and adjusts the excore nuclear instrument channel if the absolute difference between the incore and excore AFD is ≥ ~~3%~~ 1.5% AFD.

Each nuclear instrument channel is calibrated to an average weighted peripheral AFD, which accounts for the fact that neutron leakage from the peripheral fuel assemblies nearest each excore detector will have the largest effect on the channel response. This calibration method reduces the effect of changes in the radial power distribution, caused by either burnup or control rod motion, on the channel AFD calibration. The calibration method is consistent with the development of the f(ΔI) penalty functions for the overpower ΔT and overtemperature ΔT functions, which are made a function of the same average weighted peripheral AFD (i.e., the AFD used in determining the f(ΔI) penalty is calculated using the same radial weighting factors as are used to calibrate the excore detector nuclear instrument channels). The incore AFD used as the basis for comparison when performing SR 3.3.1.4 is also calculated in the same weighted peripheral manner.

If the absolute difference is ≥ ~~3%~~ 1.5% AFD the nuclear instrument channel is still OPERABLE, but must be readjusted. If the nuclear instrument channel cannot be properly readjusted, the channel is declared inoperable. This surveillance is performed to verify the f(ΔI) input to the overpower ΔT and overtemperature ΔT functions.

Two Notes modify SR 3.3.1.4. The first Note indicates that the excore nuclear instrument channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ ~~3%~~ 1.5% AFD. * * *

* * *

Technical Specification Bases 3.4.4, Reactor Coolant System (RCS) Loops, are revised as follows:

BASES

APPLICABLE SAFETY ANALYSES

MODES 1 and 2

* * *

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 100% RATED THERMAL POWER (RTP). This is the design overpower condition for two RCS loop operation. ~~The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds possible instrumentation errors.~~ The DNBR limit defines a locus of pressure and temperature points which result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

* * *

Technical Specification Bases 3.7.1, Main Steam Safety Valves (MSSVs), are revised as follows:

BASES

APPLICABLE SAFETY ANALYSES

* * *

* * *

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux ~~High~~ - High Positive Rate setpoint is reached. * * * The FSAR Section 15.4.2 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The FSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the ~~Power Range Neutron Flux-High~~ Overpower ΔT reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the ~~Power Range Neutron Flux-High~~ Overpower ΔT setpoints to an appropriate value.

* * *

ACTIONS

* * *

A.1 and A.2

With one or both steam generators with one or more MSSVs inoperable for opening, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

* * *

* * * The following equation is used to determine the maximum allowable power level for continued operation with inoperable MSSVs.

Maximum NSSS Power ≤ (100/Q) (W_s h_{fg} N) / K

Where:

* * *

To determine the Table 3.7.1-1 Maximum Allowable Power, the Maximum NSSS Power calculated using the equation above is reduced by 9% RTP to account for ~~Nuclear Instrument System~~ Overpower ΔT trip channel uncertainties.

* * *