Union Electric
Callaway Plant

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August 17, 2006

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

ULNRC-05322

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC CO.
APPLICATION FOR AMENDMENT TO
FACILITY OPERATING LICENSE NPF-30
RELOCATION OF TECHNICAL SPECIFICATION CYCLE-SPECIFIC PARAMETERS TO THE CORE OPERATING LIMITS REPORT
ADOPTION OF INDUSTRY TRAVELERS TSTF-339 AND TSTF-363

AmerenUE herewith transmits an application for amendment to Facility Operating License Number NPF-30 for the Callaway Plant.

The proposed changes will relocate cycle-specific parameter limits from Technical Specifications (TS) 2.1.1, 3.3.1, and 3.4.1 to the CORE OPERATING LIMITS REPORT (COLR) for Callaway Plant. The justification to implement the expansion of the COLR is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report." Additionally, TS 5.6.5 will be revised to add several topical reports and allow all cited topical reports to be identified by title and number only. The proposed changes will allow AmerenUE the flexibility to enhance plant operating margin and core design margin, subject to the requirements of 10 CFR 50.59, without the need for cycle-specific license amendment requests. The changes are consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-339, Revision 2, "Relocate TS Parameters to COLR" and TSTF-363, Revision 0, "Relocate Topical Report References in ITS 5.6.5, COLR."

The proposed changes to TS 2.1.1, "Reactor Core Safety Limits," will relocate the Reactor Core Safety Limits figure to the COLR and add new Safety Limits for departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature. The proposed changes to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," will relocate the Overtemperature ΔT and Overpower ΔT setpoint parameters (nominal RCS average temperature, nominal RCS operating pressure, K values, dynamic compensation time constants (τ values), and the breakpoint and slope values

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for the f (ΔI) penalty functions) to the COLR. The proposed changes to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," will relocate the pressurizer pressure and RCS average temperature values to the COLR. The proposed changes to TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," will add several topical reports and allow all cited topical reports to be identified by title and number only.

NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 4, 1988, provides guidance to licensees for the relocation of cycle-specific variables from the TS and requires that the values of these variables are included in a COLR and are determined using NRC-approved methodologies which are referenced in the TS. The proposed changes herein meet those criteria. The proposed changes will eliminate the need for future license amendments to change these relocated TS values unless an unapproved methodology were to be used in a future change.

Attachments 1 through 5 provide the Evaluation, Markup of Technical Specifications, Retyped Technical Specifications, Proposed Technical Specification Bases changes, and Draft COLR Markups, respectively, in support of this amendment request. Attachments 4 and 5 are provided for information only. Final Bases changes will be processed under our program for updates per TS 5.5.14, "Technical Specifications Bases Control Program," at the time this amendment is implemented. Final changes to the COLR will be submitted to the NRC per the update process covered by TS 5.6.5.d. No other commitments are contained in this amendment application.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

The Callaway Onsite Review Committee and a subcommittee of the Nuclear Safety Review Board have reviewed and approved the attached licensing evaluations and have approved the submittal of this amendment application.

AmerenUE requests approval of this LAR prior to March 1, 2007. AmerenUE further requests that the license amendment be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance.

In accordance with 10 CFR 50.91, a copy of this amendment application is being provided to the designated Missouri State official. If you have any questions on this amendment application, please contact me at (573) 676-8659, or Mr. Dave Shafer at (314) 554-3104.

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

Very truly yours,

Executed on: August 17, 2006

Keith D. Young

Keith D. Young
Manager, Regulatory Affairs

Attachments

1 - Evaluation
2 - Markup of Technical Specifications
3 - Retyped Technical Specifications
4 - Proposed Technical Specification Bases Changes (for information only)
5 - Draft COLR Markups (for information only)

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EVALUATION

EVALUATION

1.0 DESCRIPTION

The proposed changes will relocate cycle-specific parameter limits from Technical Specifications (TS) 2.1.1, 3.3.1, and 3.4.1 to the CORE OPERATING LIMITS REPORT (COLR) for Callaway Plant. The justification to implement the expansion of the COLR is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," (Reference 1). NRC approved this topical report in Reference 2. Additionally, TS 5.6.5 will be revised to add several topical reports and to allow all cited topical reports to be identified by title and number only. The proposed changes will allow AmerenUE the flexibility to enhance plant operating margin and core design margin, subject to the requirements of 10 CFR 50.59, without the need for cycle-specific license amendment requests in most cases. The changes are consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-339, Revision 2, "Relocate TS Parameters to COLR" (Reference 3) and TSTF-363, Revision 0, "Relocate Topical Report References in ITS 5.6.5, COLR" (Reference 4).

2.0 PROPOSED CHANGES

The proposed changes to TS 2.1.1, "Reactor Core Safety Limits," will relocate the Reactor Core Safety Limits figure to the COLR and add new Safety Limits for departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature. The proposed changes to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," will relocate the Overtemperature ΔT and Overpower ΔT setpoint parameters (nominal RCS average temperature, nominal RCS operating pressure, K values, dynamic compensation time constants (τ values), and the breakpoint and slope values for the f (ΔI) penalty functions) to the COLR. The proposed changes to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," will relocate the pressurizer pressure and RCS average temperature values to the COLR. TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," will be revised to add several topical reports and to allow all cited topical reports to be identified by title and number only.

<u>Reactor Core Safety Limits</u>

TS Figure 2.1.1-1, "Reactor Core Safety Limits," will be relocated to the COLR. In addition, TS 2.1.1 will be revised to read: "...shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The design limit departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.22 for transients analyzed using the revised thermal

design procedure (RTDP) methodology and the WRB-2 DNB correlation. For non-RTDP transients analyzed using the standard thermal design procedure, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit ($\geq$ 1.17 for WRB-2, $\geq$ 1.30 for W-3).

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWd/MTU of burnup."

Overtemperature (OT) ΔT and Overpower (OP) ΔT Setpoint Parameters

In TS Table 3.3.1-1, Note 1 (OTΔT) and Note 2 (OPΔT) establish several setpoint parameters. These values are as follows:

- T' and T" for nominal T_{avg} at rated thermal power for OTΔT and OPΔT, respectively;
- P' for nominal RCS operating pressure for OTΔT;
- K_1 through K_3 for OTΔT;
- K_4 through K_6 for OPΔT;
- Dynamic compensation time constants (τ_1 through τ_7 values for OTΔT and OPΔT, as applicable); and
- Breakpoint and slope values for the f (ΔI) penalty functions.

All of these values will be relocated from the TS to the COLR and replaced by an asterisk. A new sentence will be added to Note 1 and Note 2 that reads: "The values denoted with * are specified in the COLR."

RCS Pressure and Temperature DNB Parameters

The TS 3.4.1 LCO limits specified for pressurizer pressure and RCS average temperature will be relocated to the COLR. LCO 3.4.1.a will be revised to read: "Pressurizer pressure is greater than or equal to the limit specified in the COLR." LCO 3.4.1.b will be revised to read: "RCS average temperature is less than or equal to the limit specified in the COLR." SR 3.4.1.1 will be revised to read: "Verify pressurizer pressure is greater than or equal to the limit specified in the COLR." SR 3.4.1.2 will be revised to read: "Verify RCS average temperature is less than or equal to the limit specified in the COLR."

COLR Analytical Methods

TS 5.6.5.a will be revised to correct punctuation and add the following:

8. Reactor Core Safety Limits Figure for Specification 2.1.1,

9. Overtemperature ΔT and Overpower ΔT Setpoint Parameters for Specification 3.3.1, and

10. Reactor Coolant System Pressure and Temperature DNB Limits for Specification 3.4.1.

TS 5.6.5.b identifies the approved topical reports and analytical methods used to determine the core operating limits. This section will be revised to delete the specific revision numbers and approval dates for analytical methods 1-3 and revise current analytical method 4 to remove the specific review and approval details. TS 5.6.5.b will now specify the approved topical reports by number and title only consistent with Reference 4. TS 5.6.5.b will also be revised to add the following eight references:

5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."

6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."

7. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER ΔT AND THERMAL OVERTEMPERATURE ΔT TRIP FUNCTIONS."

8. WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."

9. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."

10. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."

11. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."

12. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."

Corresponding Bases markups for the TS 2.1.1, TS 3.3.1, and TS 3.4.1 changes are provided in Attachment 4 for information only. Page 5.0-20 in Attachment 3 (retyped TS pages) is included to eliminate "white space" from previous amendments and to preclude the need to retype the Chapter 5 pages that follow page 5.0-23.

3.0 BACKGROUND

Regulatory Approval History

NRC Generic Letter 88-16 (Reference 5), "Removal of Cycle-Specific Parameter Limits From Technical Specifications," provides guidance to licensees for the relocation of cycle-specific variables from the TS and requires that the values of these variables are included in a COLR and are determined in accordance with NRC-approved methodologies which are referenced in the TS. The proposed changes herein meet those criteria. The proposed changes will eliminate the need for future license amendments to change these relocated TS values unless an unapproved methodology were to be used in a future change.

WCAP-14483 was submitted to the NRC on December 1, 1995. That topical report provides the justification to support the TS changes required to expand current COLRs associated with Westinghouse plants. The NRC subsequently approved WCAP-14483 by letter dated January 19, 1999 (Reference 2).

TSTF-339, Revision 2 (Reference 3), contains generic changes to NUREG-1431, Revision 1, "Standard Technical Specifications Westinghouse Plants," based on WCAP-14483-A and was approved by the NRC on June 13, 2000.

TSTF-363, Revision 0 (Reference 4), revised TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," of NUREG-1431, Revision 1, to allow Topical Reports to be identified by number and title only. TSTF-363, Revision 0, was approved by the NRC on April 13, 2000.

Reactor Core Safety Limits

TS Figure 2.1.1-1, "Reactor Core Safety Limits," presents the limiting RCS average temperature conditions as a function of pressurizer pressure and thermal power. This figure is included in the TS to satisfy the requirements of 10 CFR 50.36 which states that "safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity." The Reactor Core Safety Limits figure provides the relationship between process variables available to the control room operators (RCS average temperature, pressurizer pressure, and thermal power level) and the DNB design basis. If a Condition I or II event were to occur, the Reactor Core Safety Limits figure could be used to determine whether or not the DNB design basis was met.

Since the Reactor Core Safety Limits figure is used in the generation of the OTΔT and OPΔT reactor trip setpoints, it contains the hot-leg boiling limits which are not true safety limits. The hot-leg boiling limits preclude saturation conditions to ensure that the

measured ΔT remains proportional to thermal power. The DNB limits of the figure are based on the DNBR safety analysis limits and assume a specific RCS total flow rate and a symmetrical reference axial power shape. Based on this figure, the gains (K_1 through K_6) of the OTΔT and OPΔT reactor trip setpoints are generated. For non-symmetrical power shapes that are more limiting than the reference axial power shape, the f_1 (ΔI) penalty function of the OTΔT trip function reduces the OTΔT trip setpoint (at Callaway the f_2 (ΔI) penalty function of the OPΔT trip function imposes no reduction on the OPΔT trip setpoint). Thus, the OTΔT and OPΔT reactor trip setpoints ensure that the Reactor Core Safety Limits figure is satisfied during a Condition I or II event and also ensure that, for non-symmetrical axial power shapes for which the Reactor Core Safety Limits figure may not be applicable, the DNB design basis is satisfied. Since the OTΔT and OPΔT reactor trip setpoints are based on the Reactor Core Safety Limits figure, the only way to violate the figure is under the postulated condition where both trains of the Reactor Trip System (RTS) do not function as designed. Operation of the RTS and the Main Steam Safety Valves (MSSVs) ensures that the DNB design basis is satisfied for any Condition I or II transient, independent of the Reactor Core Safety Limits figure.

Overtemperature ΔT and Overpower ΔT Setpoint Parameters

The OTΔT and OPΔT reactor trip functions ensure that, during any Condition I or II transient, there will be at least a 95% probability at a 95% confidence level that DNB will not occur on the limiting fuel rods. This is the DNB design basis discussed in greater detail in FSAR Section 4.4.1.1. If DNB is precluded, adequate heat transfer is assured between the fuel cladding and the reactor coolant, and damage due to inadequate cooling is prevented. In addition, these trip functions ensure that there will be at least a 95% probability at a 95% confidence level that the fuel rods experiencing the peak linear power (kW/ft) will not exceed the uranium dioxide (UO_2) melting temperature. This is the fuel temperature design basis discussed in greater detail in FSAR Section 4.4.1.2. In order to achieve this, a fuel centerline temperature limit has been established based on the melting temperature for UO_2 of 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

The OTΔT reactor trip function, in conjunction with the OPΔT reactor trip function, ensures operation within the DNB design basis and within the hot-leg boiling limits. Since both of these limits are functions of the coolant temperature, pressure, and core thermal power, the OTΔT reactor trip setpoint varies with measured vessel ΔT, RCS average temperature, and pressurizer pressure. A compensating term which is a function of axial flux difference (ΔI) is also factored into the OTΔT setpoint to account for potential axial power shape asymmetries. The setpoint scaling is consistent with the nominal full power operating conditions.

The OPΔT reactor trip function, in conjunction with the OTΔT and high neutron flux reactor trip functions, ensures operation within the fuel temperature design basis. This is accomplished through the OPΔT reactor trip function, with a trip setpoint that varies with measured vessel ΔT and RCS average temperature. Since thermal power is not precisely

proportional to vessel ΔT, due to the effects of changes in coolant density and heat capacity, a compensation term which is a function of vessel average temperature is factored into the calculated OPΔT trip setpoint. The setpoint scaling is consistent with the nominal full power operating conditions.

RCS Pressure and Temperature DNB Parameters

The TS limits on the DNB parameters assure that pressurizer pressure and RCS average temperature will be maintained within the limits of steady-state operation assumed in the accident analyses. These limits are consistent with the initial full power conditions considered in the Final Safety Analysis Report (FSAR) accident analyses. For Condition I and II events, where precluding DNB is the primary analysis acceptance criterion, the safety analyses have demonstrated that the DNB design basis is satisfied, assuming that the plant is operating in compliance with the TS DNB parameter limits prior to the initiation of the event.

COLR Analytical Methods

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as cited in TS 5.6.5.b.

4.0 TECHNICAL ANALYSIS

The justification to implement the expansion of the COLR is provided in Reference 1. The corresponding changes to the Standard Technical Specifications were approved in Reference 3. The justification for allowing Topical Reports to be referenced by title and number only was approved in Reference 4.

Reactor Core Safety Limits

Relocating the Reactor Core Safety Limits figure to the COLR eliminates the possibility of reaching an incorrect conclusion concerning whether or not a safety limit has been violated for a Condition I or II event. Additionally, removal of the Reactor Core Safety Limits figure prevents the possibility of misusing the Reactor Core Safety Limits figure to define an "acceptable" operating configuration that could result in the plant being placed in an unanalyzed condition. Section 4.2 of Reference 1 provides several additional reasons why this figure should be relocated to the COLR. Confirming that the RTS and the MSSVs are functioning as designed will ensure that both the DNB design basis and fuel centerline melting criteria are satisfied for any Condition I or II event. With this approach, the chance of reaching an incorrect conclusion with respect to the safety limits would be greatly reduced, if not eliminated.

As documented in the Westinghouse Owners Group (WOG) response (Reference 6) to the NRC Request for Additional Information (Reference 7) associated with Reference 1,

the Reactor Core Safety Limits figure will be relocated to the COLR; however, the requirement to operate within the limits specified in the Reactor Core Safety Limits figure will be retained in the TS. The methodology used to calculate the Reactor Core Safety Limits figure is contained in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (Reference 8). TS 5.6.5.b currently cites Reference 8 as one of the analytical methods used to determine the core operating limits. Therefore, the requirement in Reference 5 that the NRC-approved methodology used to derive the parameters in the figure be referenced in TS 5.6.5.b is already satisfied.

It is further proposed that new Safety Limits be added for the DNB design basis and the fuel centerline melting limit as these limits are criteria that must be satisfied for all Condition I and II transients. For those transients that are analyzed using the revised thermal design procedure (RTDP) methodology of WCAP-11397-P-A, "Revised Thermal Design Procedure" (Reference 9), the DNBR correlation limit is combined statistically with the independent plant system uncertainties. This results in a design limit DNBR value that establishes the new Safety Limit to meet the DNB design basis. This explains why new Safety Limit 2.1.1.1 states that the design limit DNBR shall be maintained ≥ 1.22 for transients analyzed using the RTDP methodology (note that the typical cell design limit DNBR of 1.22 in FSAR Section 4.4.1.1 is more limiting than the thimble cell design limit DNBR of 1.21) and the WRB-2 DNB correlation. For non-RTDP transients analyzed using the standard thermal design procedure (STDP), the DNB design basis is met by analyzing the plant with system uncertainties included and then assuring that the resulting DNBR value exceeds the applicable DNB correlation limit for the correlation being used. This explains why new Safety Limit 2.1.1.1 also states that, for non-RTDP transients analyzed using the STDP, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit (≥ 1.17 for WRB-2, ≥ 1.30 for W-3). New Safety Limit 2.1.1.2 requires that the peak fuel centerline temperature be maintained less than 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

FSAR Sections 4.4.1.1, "Departure from Nucleate Boling Design Basis," and 4.4.1.2, "Fuel Temperature Design Basis," provide additional information on these new Safety Limits.

Overtemperature ΔT and Overpower ΔT Setpoint Parameters

Relocation of the OTΔT and OPΔT setpoint parameters to the COLR minimizes the chance that a reload-related parameter change would require a license amendment. In addition, significant DNB and operating margin that is currently contained in these setpoint parameters may be overly conservative and unnecessarily allocated yet, absent a license amendment, this margin is unavailable to enhance plant operating flexibility, enhance the OTΔT and OPΔT setpoints, or increase the flexibility of core designs. Section 3.2 of Reference 1 provides several additional reasons why these values should be relocated to the COLR.

RCS Pressure and Temperature DNB Parameters

Relocating the DNB parameters limit values to the COLR allows the flexibility to utilize available margin to increase cycle operating margins and improve core reload designs without the requirement of cycle-specific license amendments. The relocation of these two DNB parameters to the COLR results in a more complete COLR containing not only cycle-specific core reload-related parameters, but also cycle-specific operating condition parameters. Thus the safety analyses could credit the actual cycle-specific operating condition in the same way that the core reload designs currently do. The COLR and safety analyses will more closely reflect the cycle-specific conditions for which the plant control and protection systems are set for a given cycle. Section 2.2 of Reference 1 provides several additional reasons why these two DNB parameters should be relocated to the COLR.

Even though the approved changes in References 1 and 3 indicate that the RCS total flow rate can be added to the COLR, the minimum limit for RCS total flow rate of ≥ 382,630 gpm (based on the NRC-approved analysis for the replacement steam generators) must be retained in TS 3.4.1. As documented in the NRC Request for Additional Information (RAI) in Reference 7, a "change in RCS flow from cycle-to-cycle is an indication of a physical change to the plant that should be reviewed by the staff." In order to comply with the Westinghouse Owners Group (WOG) response to RAI #2 (Reference 6), the minimum limit for RCS total flow rate will be retained in TS 3.4.1 to assure that a lower flow rate than that reviewed by the NRC will not be allowed. Thus, there is no reason to adopt this portion of the TS changes in Reference 3 since dual residency for the location of the RCS total flow rate limit would only serve to promote confusion on the review requirements associated with making changes to this value.

COLR Analytical Methods

NRC Generic Letter 88-16 (Reference 5) allows removal of cycle-specific variables from the TS and requires that the values of these variables are included in a COLR and are determined with NRC reviewed and approved methodologies which are referenced in the TS. TS 5.6.5.b identifies the approved topical reports for the analytical methods used to determine the core operating limits. TS 5.6.5.b will be revised to specify the approved topical reports by number and title only consistent with TSTF-363, Revision 0 (Reference 4). By letter dated December 15, 1999 (Reference 10), the NRC indicated that it is acceptable for the references to Topical Reports in TS Section 5.6.5 to give the Topical Report number and title only, as long as the complete citation is given in the COLR. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the COLR without having to submit an amendment to the facility operating license every time the Topical Report is revised. The COLR will provide specific information identifying the particular approved Topical Report used to determine the core limits for the current operating cycle. The proposed change to cite only the Topical Report number and title in TS 5.6.5.b does not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved

by the NRC.

Implementation of revisions to Topical Reports, which would typically occur during reload design verifications, must still be reviewed in accordance with 10 CFR 50.59 and, where required pursuant to subpart 10 CFR 50.59(c)(2)(viii), receive prior NRC review and approval.

Reactor Core Safety Limits

The Reactor Core Safety Limits figure will be relocated to the COLR; however, the requirement to operate within the limits specified in the Reactor Core Safety Limits figure will be retained in the TS. The methodology used to calculate the Reactor Core Safety Limits figure is contained in WCAP-9272-P-A (Reference 8), already cited in TS 5.6.5.b. The acceptability of citing this topical report is discussed on page 4 of the NRC Safety Evaluation attached to Reference 2.

Two new Safety Limits will be added to TS 2.1.1 for the DNB design basis and the fuel temperature design basis. A combination of methodology reports in addition to Reference 8, as discussed below, will be added to TS 5.6.5.b to address these new Safety Limits.

WCAP-11397-P-A (Reference 9) describes the design method employed to meet the DNB design basis, the revised thermal design procedure (RTDP). With the RTDP methodology, uncertainties in the plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor which is used to define the design limit DNBR that satisfies the DNB design criterion.

WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" (Reference 11), describes the three-dimensional subchannel VIPRE-01 code that has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core and hot channels. The VIPRE core model as approved by the NRC is used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core under all expected operating conditions.

The NRC-approved PAD code is the principal design tool for demonstrating that the fuel temperature design basis is met. PAD iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power. A combination of topical reports (WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations" (Reference 12), and WCAP-15063-P-A, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)" (Reference 13)) describes this methodology. WCAP-12610-P-A is already referenced in

TS 5.6.5.b. The other topical reports will be added to TS 5.6.5.b.

Overtemperature (OT) ΔT and Overpower (OP) ΔT Setpoint Parameters

Westinghouse WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions" (Reference 14), is the NRC reviewed and approved methodology topical report that is used to meet the analytical and design basis for the LCO 3.3.1 OTΔT and OPΔT setpoint parameters being relocated to the COLR. Reference 14 will be added as a referenced analytical methodology report in TS Section 5.6.5.b. The acceptability of citing this topical report is discussed on page 2 of the NRC Safety Evaluation attached to Reference 2.

RCS Pressure and Temperature DNB Parameters

References 8, 9, and 11 (discussed above) provide the NRC-approved methodology reports associated with demonstrating that the DNB design basis is met. These reports will be cited in TS 5.6.5.b to support the relocation of these DNB parameters to the COLR. The acceptability of citing Reference 8 is discussed on page 3 of the NRC Safety Evaluation attached to Reference 2.

Additional Topical Reports Added To TS 5.6.5.b

Three additional topical reports are also being added to TS 5.6.5.b due to their significance in providing methodologies for parameters being relocated to the COLR in this amendment application as well for parameters already found in the COLR.

WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code" (Reference 15), and WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" (Reference 16) will be added to TS 5.6.5.b. ANC is an advanced nodal code capable of two-dimensional and three-dimensional calculations. In this design, ANC is employed as the reference model for all safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC is used to validate one-and two dimensional results and to provide information about radial (x-y) peaking factors as a function of axial position. It has the capability of calculating discrete pin powers from the nodal information as well. PHOENIX-P is a two-dimensional, multi-group transport theory code which utilizes a 70 energy-group cross-section library. It provides the capability for cell lattice modeling on an assembly level. In this design, PHOENIX-P is used to provide homogenized, two-group cross sections for nodal calculations and feedback models. It is also used in a special geometry to generate appropriately weighted constants for the baffle/reflector regions.

WCAP-13524-P-A, Revision 1-A, "APOLLO: A One Dimensional Neutron Diffusion Theory Program" (Reference 17), is a two-group, one-dimensional diffusion-depletion

code. It uses cross sections generated by a radial averaging of the corresponding three-dimensional model cross sections and is used as a one-dimensional axial model. Thermal feedback is included in the calculational models. The axial model is used for computing axial power distributions, differential rod worths, and control rod operating limits (insertion limits, return-to-power limits), etc.

Exceptions to Previously Approved TS Changes

The new DNB design basis Safety Limit 2.1.1.1 in Reference 3 assumes that all plant transient analyses use the same DNBR methodology, although the actual limit in Reference 3 is bracketed requiring each licensee to insert their plant-specific limit(s). The Reference 3 format does not work for Callaway where most of the analyses use the RTDP, but there remain some analyses that use the STDP (see page 8 above and FSAR Section 4.4.1.1). The proposed Safety Limit 2.1.1.1 satisfies Conclusion 3 in Section 4 of the NRC Safety Evaluation attached to Reference 2, i.e., we are specifying the Callaway-specific fuel DNB design basis for new Safety Limit 2.1.1.1.

As stated above on page 9, the change in Reference 3 regarding the RCS total flow rate limit in LCO 3.4.1 is not included in this amendment application. If a minimum flow rate must be retained in LCO 3.4.1 per References 2, 6, and 7, there is nothing to be gained by adding an RCS total flow rate value to the COLR and future confusion can be avoided over the change process required to revise this value (10 CFR 50.90 for LCO 3.4.1 vs. 10 CFR 50.59 for the COLR). Since this amendment application does not revise LCO 3.4.1.c, there likewise are no changes to SR 3.4.1.3 or SR 3.4.1.4.

5.0 REGULATORY SAFETY ANALYSIS

This section addresses the standards of 10 CFR 50.92 as well as the applicable regulatory requirements and acceptance criteria.

The proposed changes will relocate cycle-specific parameter limits from Technical Specifications (TS) 2.1.1, 3.3.1, and 3.4.1 to the CORE OPERATING LIMITS REPORT (COLR) for Callaway Plant. The proposed changes to TS 2.1.1, "Reactor Core Safety Limits," will relocate the Reactor Core Safety Limits figure to the COLR and add new Safety Limits for departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature. The proposed changes to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," will relocate the Overtemperature ΔT and Overpower ΔT setpoint parameters (nominal RCS average temperature, nominal RCS operating pressure, K values, dynamic compensation time constants (τ values), and the breakpoint and slope values for the f (ΔI) penalty functions) to the COLR. The proposed changes to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," will relocate the pressurizer pressure and RCS average temperature values to the COLR. TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," will be revised to add several topical reports and allow all cited topical reports to be identified by title and number only.

5.1 No Significant Hazards Consideration (NSHC)

AmerenUE has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," Part 50.92(c), as discussed below:

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

Response: No

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no design changes. The design of the reactor trip system (RTS) instrumentation and engineered safety feature actuation system (ESFAS) instrumentation will be unaffected and these protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to this amendment request will be maintained.

The proposed changes will not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes will not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended functions to mitigate the consequences of an initiating event within the assumed acceptance limits.

The proposed changes are programmatic and administrative in nature. These changes do not physically alter safety-related systems nor affect the way in which safety-related systems perform their functions. Additional Safety Limits on the DNB design basis and peak fuel centerline temperature are being imposed in TS 2.1.1, "Reactor Core Safety Limits," and the Reactor Core Safety Limits figure is being relocated to the COLR. The additional Safety Limits are consistent with the values stated in the FSAR. The proposed changes do not, by themselves, alter any of the relocated parameter limits. The removal of the cycle-specific parameter limits from the TS does not eliminate existing requirements to comply with the parameter limits. TS 5.6.5.b continues to ensure that the analytical methods used to determine the core operating limits meet NRC reviewed and approved methodologies. TS 5.6.5.c, unchanged by this amendment application, will continue to ensure that applicable limits of the safety analyses are met.

The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the COLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports for Callaway Plant applications would still be reviewed in accordance with 10

CFR 50.59(c)(2)(viii) and, where required, receive prior NRC review and approval.

The cycle-specific parameter limits being transferred from the TS to the COLR will continue to be controlled under existing programs and procedures. The FSAR accident analyses will continue to be examined with respect to future changes in the cycle-specific parameters using NRC reviewed and approved reload design methodologies, ensuring that the evaluation of new reload designs under 10 CFR 50.59 is bounded by previously accepted analyses.

All accident analysis acceptance criteria will continue to be met with the proposed changes. The proposed changes will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR. The applicable radiological dose acceptance criteria will continue to be met.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

There are no proposed design changes nor are there any changes in the method by which any safety-related plant SSC performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No equipment performance requirements will be affected. The proposed changes will not alter any assumptions made in the safety analyses.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

The proposed amendment will not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

Relocation of cycle-specific parameter limits has no influence on, nor does it contribute in any way to, the possibility of a new or different kind of accident. The relocated cycle-specific parameter limits will continue to be calculated using the NRC reviewed and approved methodologies. The proposed changes do not alter assumptions made in the safety analyses. Operation within the core operating limits will continue to be observed.

Therefore, the proposed changes do not create the possibility of a new or different accident from any accident previously evaluated.

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

Response: No

There will be no effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The applicable radiological dose consequence acceptance criteria will continue to be met.

The proposed changes do not eliminate any surveillances or alter the frequency of surveillances required by the Technical Specifications. The nominal RTS and ESFAS trip setpoints will remain unchanged. None of the acceptance criteria for any accident analysis will be changed.

The development of cycle-specific parameter limits for future reload designs will continue to conform to NRC reviewed and approved methodologies, and will be performed pursuant to 10 CFR 50.59 to assure that plant operation within cycle-specific parameter limits.

The proposed changes will have no impact on the radiological consequences of a design basis accident.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Conclusion:

Based on the above evaluation, AmerenUE concludes that the proposed amendment presents no 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.

5.2 Applicable Regulatory Requirements / Criteria

The regulatory guidance documents associated with this amendment application include:

- The regulatory basis for TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is to ensure core operating limits are established in accordance with NRC approved methodologies and document those limits in the COLR.

- Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," provides guidance for the removal of cycle-specific parameter limits from the TS, since processing cycle-specific limit changes was an unnecessary burden on both licensees and the NRC. The Generic Letter was intended to apply to those TS changes that were developed with NRC-approved methodologies. To support the removal of cycle-specific parameter limits, the Generic Letter recommends that cycle-specific parameter limit values be placed in a CORE OPERATING LIMITS REPORT (COLR), thereby eliminating the need for many reload license amendments. The COLR would be submitted to the NRC to allow continued trending of information even though NRC approval of these limits would not be required.

- 10 CFR 50.36(c)(5) requires that the TS include a category called "Administrative Control," that contains the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

There are no changes being proposed in this amendment application such that commitments to the regulatory guidance documents above would come into question. The evaluations documented above confirm that Callaway Plant will continue to comply with all applicable regulatory requirements.

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

6.0 ENVIRONMENTAL CONSIDERATION

AmerenUE has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.1 REFERENCES

1. D. S. Huegel, J. D. Andrachek, and C. E. Morgan, "Generic Methodology for Expanded Core Operating Limits Report," WCAP-14483-A, January 1999.

2. T. H. Essig (NRC) Letter to A. P. Drake (WOG), "Acceptance For Referencing Of Licensing Topical Report WCAP-14483, 'Generic Methodology for Expanded Core Operating Limits Report,' (TAC No. M94338)," January 19, 1999.

3. TSTF-339, Revision 2, "Relocate TS Parameters to COLR."

4. TSTF-363, Revision 0, "Relocate Topical Report References in ITS 5.6.5, COLR."

5. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988.

6. Letter from L. F. Liberatori Jr. (WOG) Response to NRC Document Control Desk, Westinghouse Owners Group Response to NRC Request for Additional Information on WCAP-14483, "Generic Methodology for Expanded Core Operating Limits Report," (Non-Proprietary), (MUHP-1009), OG-98-118 dated November 25, 1998.

7. P. C. Wen (NRC) Request for Additional Information to A. P. Drake (WOG), September 2, 1998.

8. S. L. Davidson, et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

9. A. J. Friedland and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

10. S. A. Richards (NRC) Letter to J. F. Malley (Siemens Power Corporation), "Acceptance For Siemens References to Approved Topical Reports in Technical Specifications," December 15, 1999.

11. Y. X. Sung, et. al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A, October 1999.

12. R. A. Weiner, et. al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

13. J. P. Foster, et. al., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1, with Errata, July 2000.

14. J. A. Fici, et. al., "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," WCAP-8745-P-A, September 1986.

15. Y. S. Liu, A. Meliksetian, J. A. Rathkopf, D. C. Little, F. Nakano, M. J. Poplaski, "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

16. T. Q. Nguyen, et. al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June, 1988.

17. M. B. Yarbrough, et. al., "APOLLO: A One Dimensional Neutron Diffusion Theory Program," WCAP-13524-P-A, Revision 1-A, September 1997.

7.2 PRECEDENT

Although several licensees have adopted References 1-4, and no particular plant's implementation may look the same as another's due to the likelihood of plant-specific analyses, Wolf Creek's Amendment No. 144 dated March 28, 2002 is cited as a precedent for Callaway. Other than the two items discussed on page 12 above under Exceptions to Previously Approved TS Changes, and the listing of plant-specific methodology reports in TS 5.6.5.b, the changes contained in this amendment application are similar to those contained in Wolf Creek's amendment application ET 01-0008 dated April 3, 2001, as supplemented on October 22, 2001 and December 18, 2001 (ET 01-0030 and ET 01-0036), and March 7, 2002 (GC 02-0011). To the extent practicable, the RAI responses in the October 22 and December 18, 2001 supplements have been factored into this amendment application. The OTΔT Note 1 mathematical sign error corrected in the March 7, 2002 supplement does not apply to this amendment application.

ATTACHMENT 2

MARKUP OF TECHNICAL SPECIFICATIONS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits ~~SLs~~ specified in ~~Figure 2.1.1-1.~~ INSERT A

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

INSERT A

the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The design limit departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.22 for transients analyzed using the revised thermal design procedure (RTDP) methodology and the WRB-2 DNB correlation. For non-RTDP transients analyzed using the standard thermal design procedure, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit ($\geq$ 1.17 for WRB-2, $\geq$ 1.30 for W-3).

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWd/MTU of burnup.

Deleted from TS;
moved to COLR.

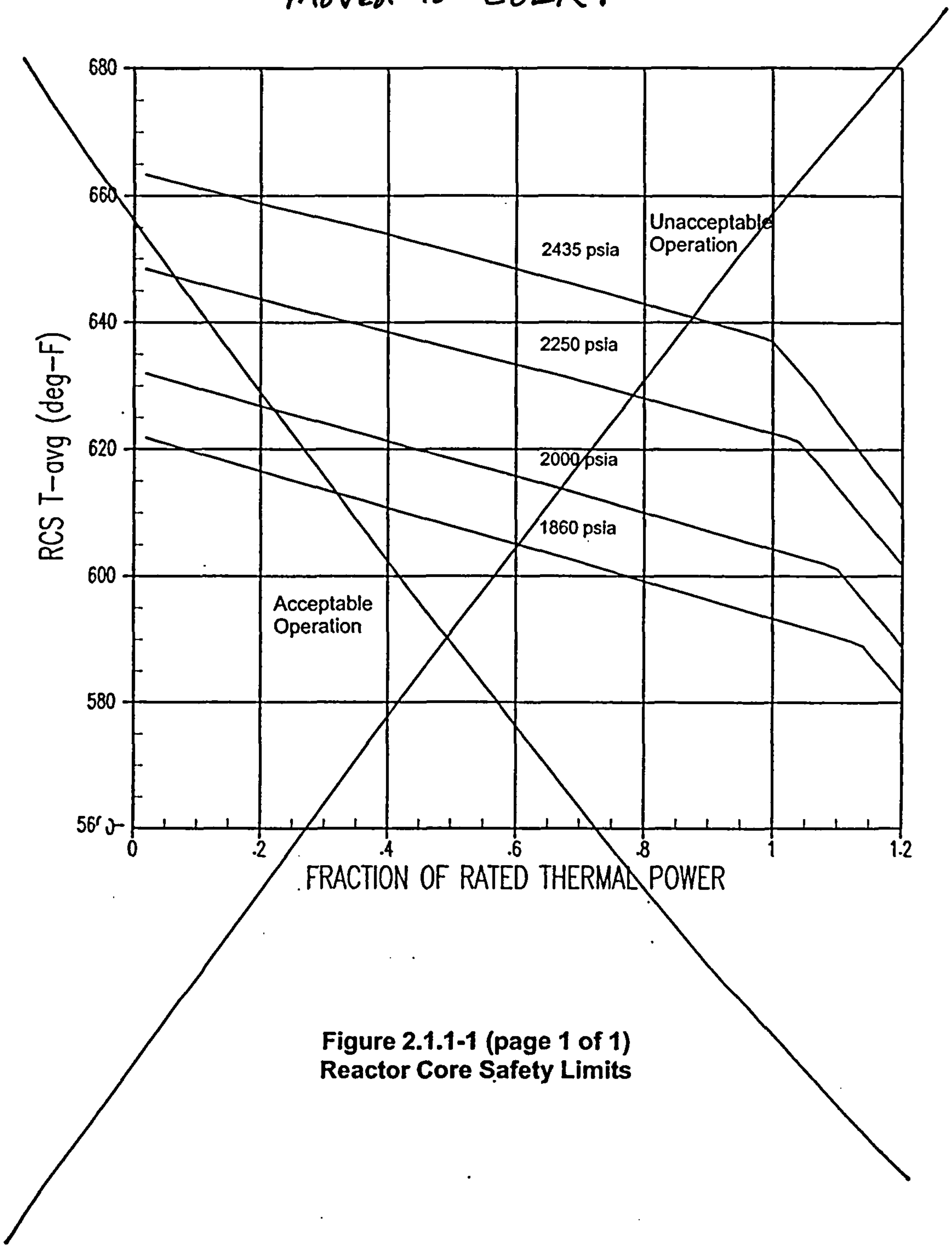

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following setpoint by more than 1.23% of ΔT span (1.85% RTP).

$$\Delta T\frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_O\left\{K_1 - K_2\left[\frac{(1+\tau_4 s)}{(1+\tau_5 s)}T\frac{1}{(1+\tau_6 s)} - T'\right] + K_3(P-P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ ~~585.0~~ * °F.

P is the measured pressurizer pressure, psig. *
P′ is the nominal RCS operating pressure = ~~2235~~ psig.

K_1 = ~~1.1050~~ * | K_2 = ~~0.0254~~ * /°F | K_3 = ~~0.00110~~ * /psig
$\tau_1 \ge$ ~~8~~ * sec | $\tau_2 \le$ ~~3~~ * sec | τ_3 = ~~0~~ * sec
$\tau_4 \ge$ ~~28~~ * sec | $\tau_5 \le$ ~~4~~ * sec | τ_6 = ~~0~~ * sec

$f_1(\Delta I)$ =
- ~~-0.0325~~ * {~~21%~~ * + $(q_t - q_b)$} when $q_t - q_b$ < ~~-21%~~ * RTP
- 0% of RTP * when ~~-21%~~ RTP ≤ $q_t - q_b$ ≤ ~~8%~~ * RTP
- ~~0.02979~~ * {$(q_t - q_b)$ - ~~8%~~ *} when $q_t - q_b$ > ~~8%~~ * RTP

where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

INSERT B

INSERT B

The values denoted with * are specified in the COLR.

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following setpoint by more than 1.21% of ΔT span (1.82% RTP).

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_O \left\{ K_4 - K_5 \frac{(\tau_7 s)}{(1+\tau_7 s)}\left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{(1+\tau_6 s)} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T" is the nominal T_{avg} at RTP, ≤ ~~585.3~~ * °F.

K_4 = ~~1.1073~~ *

K_5 = ~~0.02~~ * /°F for increasing T_{avg}
* ~~0~~/°F for decreasing T_{avg}

K_6 = ~~0.0015~~ * /°F when T > T"
* ~~0~~/°F when T ≤ T"

τ_1 ≥ ~~8~~ * sec
τ_2 ≤ ~~3~~ * sec
τ_3 = ~~0~~ * sec
τ_6 = ~~0~~ * sec
τ_7 ≥ ~~10~~ * sec

$f_2(\Delta I)$ = ~~0% RTP for all ΔI.~~ *

INSERT B

INSERT B

The values denoted with * are specified in the COLR.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ~~≥ 2223 psig;~~ is INSERT C;

b. RCS average temperature ~~≤ 590.1°F; and~~ is INSERT D; and

c. RCS total flow rate ≥ 382,630 gpm.

APPLICABILITY: MODE 1.

-----------------------------------NOTE-----------------------------------

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

INSERT C

greater than or equal to the limit specified in the COLR

INSERT D

less than or equal to the limit specified in the COLR

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ~~≥ 2223 psig.~~ INSERT C.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ~~≤ 590.1°F.~~ INSERT D.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 382,630 gpm.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Calculated rather than verified by precision heat balance when performed prior to THERMAL POWER exceeding 75% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 382,630 gpm.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP *AND* 18 months

INSERT C

greater than or equal to the limit specified in the COLR

INSERT D

less than or equal to the limit specified in the COLR

5.6.2 Annual Radiological Environmental Operating Report (continued)

reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year. in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Not used.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient limits in Specification 3.1.3,

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference Limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor, $F_Q(Z)$ F_Q^{RTP}, K(Z), W(Z) and F_Q Penalty Factors for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3.2.2~~and~~,

7. Shutdown Margin Limits for Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8~~and~~,

INSERT E

(continued)

INSERT E

8. Reactor Core Safety Limits Figure for Specification 2.1.1,

9. Overtemperature ΔT and Overpower ΔT Setpoint Parameters for Specification 3.3.1, and

10. Reactor Coolant System Pressure and Temperature DNB Limits for Specification 3.4.1.

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", ~~July 1985 (W Proprietary).~~

2. WCAP-10216-P-A, ~~REV. 1A,~~ "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION," ~~February 1994 (W Proprietary).~~

3. WCAP-10266-P-A, ~~REV. 2,~~ "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," ~~March 1987 (W Proprietary).~~

INSERT F

4. ~~NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report' (TAC NO 77268)," and September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC No. M86416)" (WCAP-12610-P-A).~~

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

INSERT F

4. WCAP-12610-P-A, "VANTAGE+ FUEL ASSEMBLY REFERENCE CORE REPORT."

5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."

6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."

7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."

8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."

9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER ΔT AND THERMAL OVERTEMPERATURE ΔT TRIP FUNCTIONS."

10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."

11. WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."

12. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."

ATTACHMENT 3

RETYPED TECHNICAL SPECIFICATIONS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The design limit departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.22 for transients analyzed using the revised thermal design procedure (RTDP) methodology and the WRB-2 DNB correlation. For non-RTDP transients analyzed using the standard thermal design procedure, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit (≥ 1.17 for WRB-2, ≥ 1.30 for W-3).

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWd/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

TABLE 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature T

The Overtemperature ΔT Function Allowable Value shall not exceed the following setpoint by more than 1.23% of T span (1.85% RTP).

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_o \left\{ K_1 - K_2\left[\frac{(1+\tau_4 s)}{(1+\tau_5 s)} T \frac{1}{1+\tau_6 s} - T'\right] + K_3(P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_o is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ *°F.

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure = * psig.

K_1 = *	K_2 = */°F	K_3 = */psig
$\tau_1 \ge$ * sec	$\tau_2 \le$ * sec	τ_3 = * sec
$\tau_4 \ge$ * sec	$\tau_5 \le$ * sec	τ_6 = * sec

$f_1(\Delta I)$ =	* {*% + $(q_t - q_b)$}	when $q_t - q_b$ < * %RTP
	0% of RTP	when * %RTP ≤ $q_t - q_b$ ≤ * %RTP
	* {$(q_t - q_b)$ - *}	when $q_t - q_b$ > * %RTP

where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

The values denoted with * are specified in the COLR.

TABLE 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower T

The Overpower ΔT Function Allowable Value shall not exceed the following setpoint by more than 1.21% of T span (1.82% RTP).

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_o\left\{K_4 - K_5\frac{(\tau_7 s)}{(1+\tau_7 s)}\left(\frac{1}{1+\tau_6 s}\right)T - K_6\left[T\frac{1}{(1+\tau_6 s)} - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_o is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature,°F.
T" is the nominal T_{avg} at RTP, ≤ *°F.

K_4 = *	K_5 = */°F for increasing T_{avg} */°F for decreasing T_{avg}	K_6 = */°F when T > T" */°F when T ≤ T"
$\tau_1 \ge$ * sec	$\tau_2 \le$ * sec	τ_3 = * sec
τ_6 = * sec	$\tau_7 \ge$ * sec	

$f_2(\Delta I)$ = *

The values denoted with * are specified in the COLR.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure is greater than or equal to the limit specified in the COLR;

b. RCS average temperature is less than or equal to the limit specified in the COLR; and

c. RCS total flow rate ≥ 382,630 gpm.

APPLICABILITY: MODE 1.

--- NOTE ---

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1	Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 382,630 gpm.	12 hours
SR 3.4.1.4	-------------------------------- NOTE --------------------------- Calculated rather than verified by precision heat balance when performed prior to THERMAL POWER exceeding 75% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 382,630 gpm.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 18 months

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period.

The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year. in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Not used.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient limits in Specification 3.1.3,

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference Limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor, $F_Q(Z)$ F_Q^{RTP}, K(Z), W(Z) and F_Q Penalty Factors for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3.2.2,

7. Shutdown Margin Limits for Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8,

8. Reactor Core Safety Limits Figure for Specification 2.1.1,

9. Overtemperature ΔT and Overpower ΔT Setpoint Parameters for Specification 3.3.1, and

10. Reactor Coolant System Pressure and Temperature DNB Limits for Specification 3.4.1.

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:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY."

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION."

3. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE."

(continued)

5.6 Reporting Requirements

5.6.5 <u>CORE OPERATING LIMITS REPORT (COLR)</u> (continued)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT."

5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."

6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."

7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."

8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."

9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER ΔT AND THERMAL OVERTEMPERATURE ΔT TRIP FUNCTIONS."

10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."

11. WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."

12. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Cold Overpressure Mitigation System (COMS)."

b. The analytical methods used to determine the RCS pressure and temperature and COMS PORV limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NRC letter, CALLAWAY PLANT, UNIT 1 – ISSUANCE OF AMENDMENT RE: PRESSURE TEMPERATURE LIMITS REPORT (TAC NOS. MA5631 and MA7287), dated March 24, 2000.

2. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January, 1996".

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used.

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not used.

(continued)

ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(for information only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which requires that the minimum departure from nucleate boiling ratio (DNBR) of the limiting rod during Condition I and II events is greater than or equal to the DNBR design limits. In meeting this design basis, for Revised Thermal Design Procedure (RTDP) analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation (WRB-2) predictions are combined statistically to obtain the overall DNBR uncertainty factor. This DNBR uncertainty factor is used to define the design limit DNBR, which corresponds to a 95% probability with 95% confidence that DNB will not occur on the limiting fuel rods during Condition I and II events. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their normal values. The design limit DNBR values are 1.21 and 1.22 for thimble and typical cells, respectively, for VANTAGE 5 fuel. In addition, margin has been maintained by meeting safety analysis DNBR limits of 1.55 and 1.59 for thimble and typical cells, respectively, for VANTAGE 5 fuel. The 3 design limit DNBRs are considered design bases limits for fission barriers for consideration in the 10 CFR 50.59 process. Reference 8 discusses two non-RTDP ~~three~~ transients analyzed with the W-3 DNBR correlation. INSERT H

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation

(continued)

INSERT H

(the inadvertent opening of a steam generator relief or safety valve is no longer an analyzed event) and two non-RTDP transients analyzed with the WRB-2 DNBR correlation. The correlation limits for the W-3 and WRB-2 DNBR correlations are 1.3 and 1.17, respectively.

BASES

BACKGROUND (continued)

temperature. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. Reference ~~2~~ 4 further discusses the fuel centerline temperature design basis.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Trip ~~Protection~~ System (~~RPS~~ RTS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the limiting hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System Allowable Values in Table 3.3.1-1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Protection for these reactor core SLs is provided by the proper operation of the steam generator safety valves and the following automatic reactor trip functions:

a. High pressurizer pressure trip;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Low pressurizer pressure trip;

c. Low reactor coolant system flow;

d. ~~c.~~ Overtemperature ΔT trip;

e. ~~d.~~ Overpower ΔT trip; and

f. ~~e.~~ Power Range Neutron Flux trip.

~~The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.~~

The SLs represent a design requirement for establishing the ~~RPS~~ RTS Allowable Values identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

reactor core safety limits figure

the COLR shows

limit

The ~~curves~~ provided in ~~Figure 2.1.1-1 show~~ the loci of points of THERMAL POWER, ~~RCS~~ pressurizer pressure, and average temperature below which the calculated DNBR is not less than the design DNBR value s, ~~fuel centerline temperature remains below melting,~~ the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation.

INSERT I

~~The curves are based on enthalpy rise hot channel factor limits provided in the COLR and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the equation given in the COLR.~~

~~The SL is higher than the limit calculated when the AFD is within the limits of the f1(ΔI) function of the Overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the Overtemperature ΔT reactor trip will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).~~

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation

(continued)

INSERT I

The reactor core SLs are established to preclude the violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RTS functions that the above criteria are satisfied during steady state operation, normal operating transients, and anticipated operational occurrences (AOOs). To ensure that the RTS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RTS ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

Reference 4 discusses the fuel temperature design basis. Figure 15.0-1 of Reference 2 depicts the protection provided by the Overpower ΔT reactor trip function against fuel centerline melting.

BASES

APPLICABILITY (continued)	within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Allowable Values for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.
SAFETY LIMIT VIOLATIONS	The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 10. 2. FSAR, Chapter 15. ~~3. WCAP-8746-A, March 1977.~~ ~~4. WCAP-9273-NP-A, July 1985.~~ 3. ~~5.~~ FSAR Section 4.4.1.1. 4. ~~6.~~ FSAR Section 4.4.1.2.

BASES

ACTIONS

B.1 (continued)

the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be inserted to approximately 100 steps.

Power operation may continue with one RCCA OPERABLE (i.e. trippable) but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Trip ~~Protection~~ System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic

(continued)

BASES

ACTIONS

A.1.1 (continued)

requirements. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

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 to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing 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. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints; however, for extended operations at the reduced power level, the reduced trip setpoints are required to protect against transients involving positive reactivity excursions. This is a sensitive operation that may inadvertently ~~trip~~ actuate the Reactor ~~Protection~~ Trip System.

A.2

Once actions have been taken to restore $F^N_{\Delta H}$ to within its limits per Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which

(continued)

No changes

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

5. Source Range Neutron Flux (continued)

System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide inputs to the BDMS as addressed in LCO 3.3.9, "Boron Dilution Mitigation System (BDMS)," to protect against inadvertent reactivity changes that may occur as a result of events like an uncontrolled boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature ΔT

The Overtemperature ΔT trip Function 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. The inputs to the Overtemperature ΔT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure - the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution f(ΔI) - the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limits, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced.

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

ΔT_0 and T', as used in the Overtemperature ΔT trip, represent the 100% RTP values as measured by the plant for each loop. For

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

6. Overtemperature ΔT (continued)

the startup of a refueled core until reset to actual measured values (at 90-100% RTP), ΔT_o is initially set at a value which is conservatively lower than the last measured 100% RTP ΔT_o for each loop. Setting ΔT_o and T′ to the measured value of ΔT_o and T′ normalizes each loop's Overtemperature ΔT trip to the RCS loop conditions existing at the time of measurement, thus the trip reflects the equivalent full power conditions assumed for the OTΔT trip in the accident analyses. These differences in vessel ΔT and T_{avg} can result from several factors, two of them being measured RCS loop flows greater than Minimum Measured Flow and asymmetric power distributions between quadrants. While RCS loop flows are not expected to change, radial power redistribution between quadrants may occur resulting in small changes in loop-specific vessel ΔT and T_{avg} values. Accurate determination of the loop-specific vessel ΔT and T_{avg} values are made when performing the Incore/Excore quarterly recalibration under steady state conditions (i.e., power distributions not affected by xenon or other transient conditions).

The time constants used in the lag compensation of measured ΔT (τ_3) and measured T_{avg} (τ_6) are set at 0 seconds. This setting corresponds to the 7300 NLL card values used for lag compensation of these signals. Safety analyses that credit Overtemperature ΔT for protection must account for these field adjustable lag cards as well as all other first order lag contributions (i.e., the combined RTD/thermowell response time and the scoop transport delay and thermal lag). The safety analyses use a total first order lag of less than or equal to 6 seconds.

The Overtemperature ΔT trip Function is with values as specified in the COLR. calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. The pressure and temperature signals are used for other control functions; thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power, either through automatic rod insertion or through operator action. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

(continued)

No changes

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

6. Overtemperature ΔT (continued)

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE (two-out-of-four trip logic). Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. 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. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. The Overpower ΔT trip also provides protection to mitigate the consequences of small steamline breaks, as reported in Reference 11, and the decrease in feedwater temperature event (Ref. 13). It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature - including dynamic compensation for the delays between the core and the temperature measurement system.

ΔT_0 and T", as used in the Overpower ΔT trip, represent the 100% RTP values as measured by the plant for each loop. For the startup of a refueled core until reset to actual measured values (at 90-100% RTP), ΔT_0 is initially set at a value which is conservatively lower than the last measured 100% RTP ΔT_0 for each loop. Setting ΔT_0 and T" to the measured value of ΔT_0 and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

7. Overpower ΔT (continued)

T" normalizes each loop's Overpower ΔT trip to the RCS loop conditions existing at the time of measurement, thus the trip reflects the equivalent full power conditions assumed for the OPΔT trip in the accident analyses. These differences in vessel ΔT and T_{avg} can result from several factors, two of them being measured RCS loop flows greater than Minimum Measured Flow and asymmetric power distributions between quadrants. While RCS loop flows are not expected to change, radial power redistribution between quadrants may occur resulting in small changes in loop-specific vessel ΔT and T_{avg} values. Accurate determination of the loop-specific vessel ΔT and T_{avg} values are made when performing the Incore/Excore quarterly recalibration under steady state conditions (i.e., power distributions not affected by xenon or other transient conditions).

The time constants used in the lag compensation of measured ΔT (τ_3) and measured T_{avg} (τ_6) are set at 0 seconds. This setting corresponds to the 7300 NLL card values used for lag compensation of these signals. Safety analyses that credit Overpower ΔT for protection must account for these field adjustable lag cards as well as all other first order lag contributions (i.e., the combined RTD/ thermowell response time and the scoop transport delay and thermal lag). The safety analyses use a total first order lag of less than or equal to 6 seconds.

with values as specified in the COLR.
The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE (two-out-of-four trip logic). Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

(continued)

No changes

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

7. Overpower ΔT (continued)

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure - High and - Low trips and the Overtemperature ΔT trip. The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System; thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels of Pressurizer Pressure – Low to be OPERABLE (two-out-of-four trip logic). The Trip Setpoint is ≥ 1885 psig.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure - Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than 10% of full power equivalent (P-13)). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, there is insufficient heat production to generate DNB conditions.

b. Pressurizer Pressure - High

The Pressurizer Pressure - High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the

(continued)

BASES

LCO

2, 3. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures (Wide Range) (continued)

Trip

the narrow range RTDs providing inputs to the Reactor ~~Protection~~ System. The wide range channels provide indication over a range of 0°F to 700°F. Loops 1 and 2 have hot and cold leg wide range Class 1E temperature indications in the main control room.

4. Reactor Coolant System Pressure (Wide Range)

RCS wide range pressure is a Type A, Category 1 variable provided for verification of core cooling and long term surveillance of RCS integrity.

RCS pressure is used to verify delivery of SI flow to RCS from at least one train when the RCS pressure is below the pump shutoff head. RCS pressure is also used to verify closure of manually closed pressurizer spray line valves and pressurizer power operated relief valves (PORVs).

In addition to these verifications, RCS pressure is used for determining RCS subcooling margin. RCS subcooling margin is used to determine whether to terminate SI, if still in progress, or reinitiate SI if it has been stopped. RCS pressure can also be used:

- to determine whether to terminate actuated SI or to reinitiate stopped SI;
- to determine when to reset SI and shut off low head SI;
- to manually restart low head SI;
- as reactor coolant pump (RCP) trip criteria; and
- to make a determination on the nature of the accident in progress and where to go next in the procedure.

RCS subcooling margin is also used for unit stabilization and cooldown control.

RCS pressure is also related to three decisions about depressurization. They are:

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR ~~criterion. This is the acceptance limit for the RCS DNB parameters.~~ Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution limits are satisfied per LCO 3.1.4, "Rod Group Alignment Limits;" LCO 3.1.5, "Shutdown Bank

limits (Ref. 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Insertion Limits;" LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2223 psig and the RCS average temperature limit of 590.1°F, as specified in the COLR, correspond to analytical limits of 2205 psig and 592.7°F used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. INSERT J Operating within these limits will result in meeting the DNBR ~~criterion~~ limits in the event of a DNB limited transient.

The RCS total flow rate limit contains a measurement error of 2.1% based on performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater flow venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. ~~The RTDP thermal hydraulic analyses assume an RCS flow measurement uncertainty of 2.1% and a total RCS minimum measured flow of 382,630 gpm.~~ INSERT K

Any fouling that might bias the flow rate measurement can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling. INSERT L

The ~~LCO~~ numerical values for pressure, temperature, and flow rate have been adjusted for instrument error, as discussed above.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL

(continued)

INSERT J

The limit values for pressurizer pressure and RCS average temperature are specified in the COLR to provide operating and analysis flexibility from cycle to cycle.

INSERT K

Therefore, a bias of 0.1% for undetected fouling of the feedwater flow venturi raises the nominal flow measurement allowance to 2.2%. The RTDP thermal-hydraulic analyses assume a total RCS minimum measured flow of 382,630 gpm, which is 2.2% greater than the thermal design flow of 374,400 gpm used in non-RTDP analyses.

INSERT L

greater than the 0.1% bias for undetected fouling of the feedwater flow venturi

BASES

APPLICABILITY (continued)

POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

~~Another set of limits on DNB related parameters is~~ The DNBR limits are provided in SL 2.1.1, "Reactor Core SLs," (Ref. 2). The conditions which define the DNBR limits ~~These limits~~ are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and reduce the potential for violation of the accident analysis limits.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

(continued)

No change

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. When performing a precision heat balance, the instrumentation used for determining steam pressure, feedwater temperature, and feedwater venturi Δp in the calorimetric calculations shall be calibrated within 7 days prior to performing the heat balance.

The Frequency of once after each refueling prior to THERMAL POWER exceeding 75% RTP, and 18 months reflects the importance of verifying

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.4 (continued)

flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows this SR to be performed by RCS flow calibration after each refueling prior to THERMAL POWER exceeding 75% RTP. The determination of RCS flow through a precision heat balance is not required prior to entry into MODE 1 because RCS flow indication is available (RCS flow meters), RCS flow is calculated prior to exceeding 75% RTP and the surveillance from the previous cycle is still current. The precision heat balance flow rate surveillance is typically performed after reaching full power following a refueling outage and suitable plant conditions are established.

REFERENCES

1. FSAR, Chapter 15.
2. SL 2.1.1, "Reactor Core Safety Limits (SLs)."

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Trip ~~Protection~~ System, overpressure protection for the RCS. The pressurizer safety valves are of the pop type. The valves are spring loaded and self actuated by direct fluid pressure with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are self actuating, they are considered independent components. The minimum relief capacity for each valve, 420,000 lb/hr at 2485 psig plus 3% accumulation, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves which is divided equally between the three valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 with the reactor vessel head on; however, in MODE 4 with one or more RCS cold leg temperatures ≤ 275°F, MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

The upper and lower pressure limits are based on the tolerance requirements assumed in the safety analyses. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Furthermore, for a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is positive at any power level the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. The 4 hour Completion Time for required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The completion time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor ~~protection system~~ trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provides sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If THERMAL POWER or the Power Range Neutron Flux - High trip setpoints is not reduced within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital buses. The inverters are normally powered from the station battery; however, a backup AC source provides another source of inverter output via a static switch internal to the inverter. An alternate source of power to the AC vital buses is provided from Class 1E constant voltage transformers. The station battery provides an uninterruptible power source for the instrumentation and controls for the Reactor ~~Protection~~ Trip System (~~RPS~~ RTS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in the FSAR, Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the ~~RPS~~ RTS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and

b. A worst case single failure.

Inverters satisfy Criterion 3 of the 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the ~~RPS~~ RTS and ESFAS instrumentation and controls is maintained. The four inverters (two per train) ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized.

Operable inverters require the associated vital bus to be powered by the inverter with output voltage within tolerances, and power input to the inverter from a 125 VDC station battery.

The required inverters/AC vital buses are associated with the DC electrical power subsystems (Train A and Train B) as follows:

TRAIN A		TRAIN B	
Bus NN01 energized from Inverter NN11 connected to DC bus NK01	Bus NN03 energized from Inverter NN13 connected to DC bus NK03	Bus NN02 energized from Inverter NN12 connected to DC bus NK02	Bus NN04 energized from Inverter NN14 connected to DC bus NK04

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation of the ~~RPS~~ RTS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. FSAR, Chapter 8.
2. FSAR, Chapter 6.
3. FSAR, Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor ~~Protection~~ Trip System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each AC vital bus during MODES 5 and 6 ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of

(continued)

ATTACHMENT 5

DRAFT COLR MARKUPS TO ULNRC-05228 DATED 11-4-05
(for information only)

CALLAWAY CYCLE 15

CORE OPERATING LIMITS REPORT

(Revision 0)

September 2005

Reviewed By: Charles B. Dunford 10-25-05
Approved By: [signature] / 10-25-05

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Callaway Plant Cycle 15 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The Core Operating Limits affecting the following Technical Specifications are included in this report.

3.1.1, 3.1.4, 3.1.5, 3.1.6, 3.1.8	Shutdown Margin
3.1.3	Moderator Temperature Coefficient
3.1.5	Shutdown Bank Insertion Limits
3.1.6	Control Bank Insertion Limits
3.2.1	Heat Flux Hot Channel Factor
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor
3.2.3	Axial Flux Difference
2.1.1	Reactor Core Safety Limits (SLs)
3.3.1	Reactor Trip System (RTS) Instrumentation
3.4.1	RCS Pressure and Temperature Departure from Nucleate Boiling (DNB) Limits

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the subsections which follow. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 5.6.5.

2.1 Shutdown Margin

(Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8)

2.1.1 The Shutdown Margin in MODES 1-4 shall be greater than or equal to 1.3% Δk/k.

2.1.2 The Shutdown Margin prior to blocking Safety Injection below P-11 in MODES 3 and 4 shall be greater than 0% Δk/k as calculated at 200°F.

2.1.3 The Shutdown Margin in MODE 5 shall be greater than or equal to 1.0% Δk/k.

2.2 Moderator Temperature Coefficient

(Specification 3.1.3)

2.2.1 The Moderator Temperature Coefficient shall be less positive than the limits shown in Figure 1. These limits shall be referred to as upper limit.

The Moderator Temperature Coefficient shall be less negative than -47.9 pcm/°F. This limit shall be referred to as the lower limit.

2.2.2 The MTC 300 ppm surveillance limit is -40.4 pcm/°F (all rods withdrawn, Rated Thermal Power condition).

2.2.3 The MTC 60 ppm surveillance limit is -45.5 pcm/°F (all rods withdrawn, Rated Thermal Power condition).

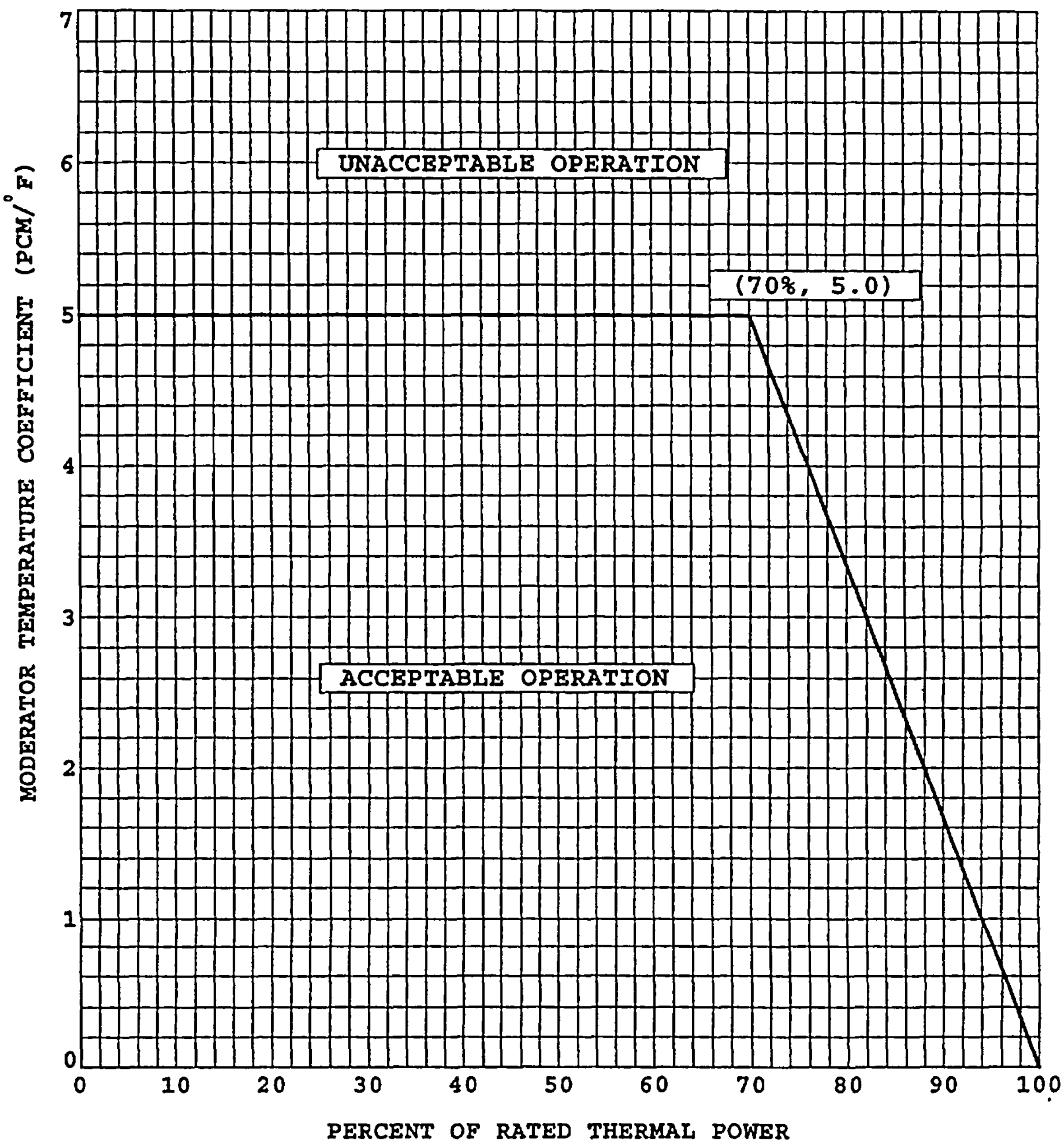

Figure 1

Callaway Cycle 15
Moderator Temperature Coefficient
Versus Power Level

2.3 Shutdown Bank Insertion Limits
(Specification 3.1.5)

The shutdown banks shall be withdrawn to at least 225 steps.

2.4 Control Bank Insertion Limits
(Specification 3.1.6)

2.4.1 Control Bank insertion limits are specified by Figure 2.

2.4.2 Control Bank withdrawal sequence is A-B-C-D. The insertion sequence is the reverse of the withdrawal sequence.

2.4.3 The difference between each sequential Control Bank position is 115 steps when not fully inserted and not fully withdrawn.

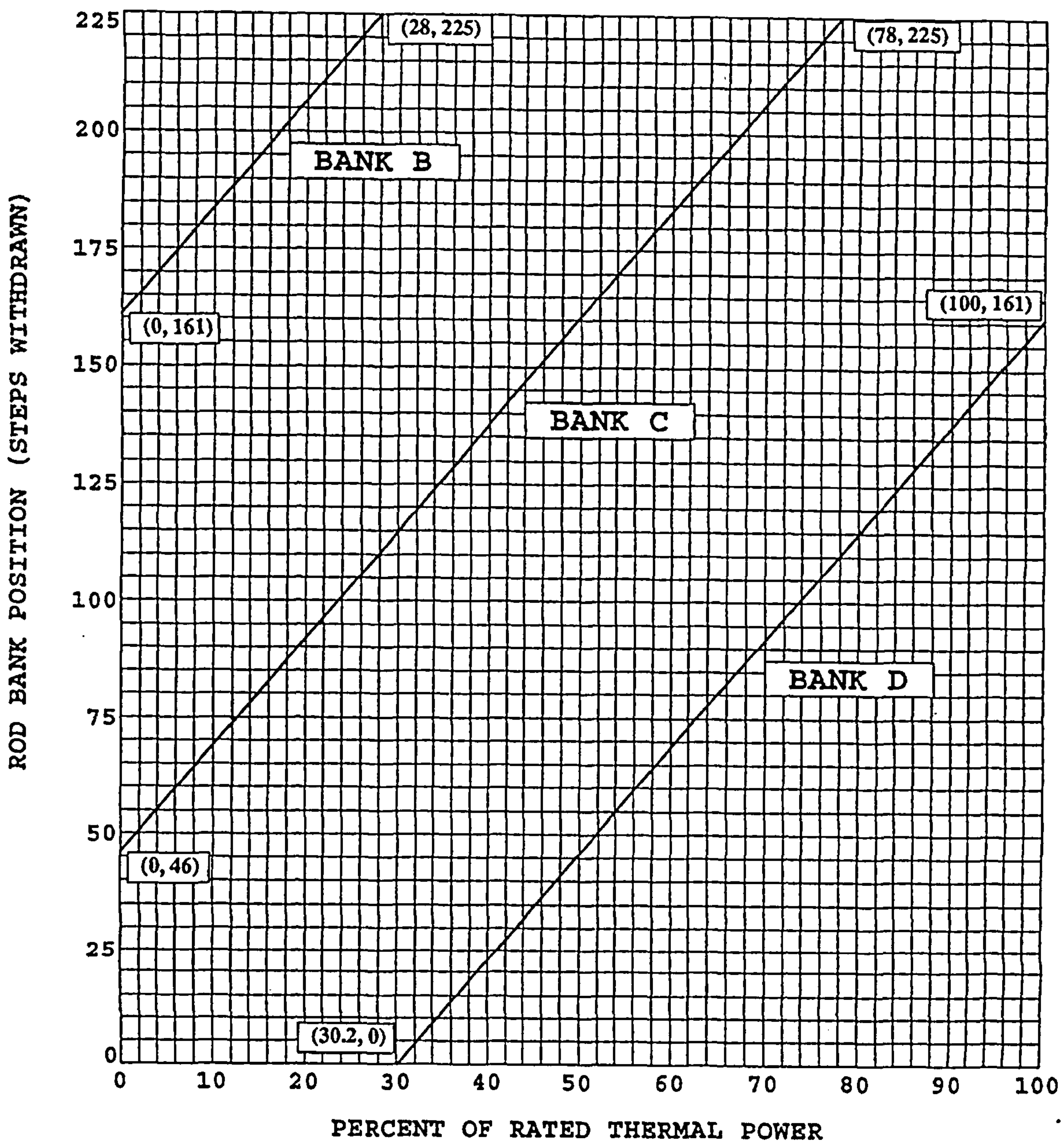

Figure 2

Callaway Cycle 15
Rod Bank Insertion Limits
Versus Rated Thermal Power - Four Loop Operation

2.5 Heat Flux Hot Channel Factor - $F_Q(Z)$
(Specification 3.2.1)

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z) \quad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} * K(Z) \quad \text{for } P \leq 0.5$$

where: $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

2.5.1 $F_Q^{RTP} = 2.50$

2.5.2 K(Z) is provided in Figure 3.

2.5.3 The W(z) functions that are to be used in Technical Specification 3.2.1 and Surveillance Requirement 3.2.1.2 for determining $F_Q^W(z)$ are shown in Table A.1.

The W(z) values have been determined for several burnups up to 18000 MWD/MTU in Cycle 15. This permits determination of W(z) at any cycle burnup up to 18000 MWD/MTU through the use of three point interpolation. For cycle burnups greater than 18000 MWD/MTU, use of 18000 MWD/MTU W(z) values without interpolation or extrapolation is conservative. The W(z) values were determined assuming Cycle 15 operates with RAOC strategy. Also included is a W(z) function that bounds the W(z) values for all Cycle 15 burnups. Use of the bounding W(z) values will be conservative for any Cycle 15 burnup; however, additional margin may be gained by using the burnup dependent W(z) values.

Table A.2 shows the burnup dependent F_Q penalty factors for Cycle 15. These values shall be used to increase $F_Q^W(z)$ when required by Technical Specification Surveillance Requirement 3.2.1.2. A 2% penalty factor should be used at all cycle burnups that are outside the range of Table A.2.

Table A.1
W(z) versus Core Height
(Top and Bottom 15% Excluded)

Height (feet)	150 MWD/MTU	4000 MWD/MTU	10000 MWD/MTU	18000 MWD/MTU	Bounding W(z)
0.00 (bottom)	1.0000	1.0000	1.0000	1.0000	1.0000
0.17	1.0000	1.0000	1.0000	1.0000	1.0000
0.33	1.0000	1.0000	1.0000	1.0000	1.0000
0.50	1.0000	1.0000	1.0000	1.0000	1.0000
0.67	1.0000	1.0000	1.0000	1.0000	1.0000
0.83	1.0000	1.0000	1.0000	1.0000	1.0000
1.00	1.0000	1.0000	1.0000	1.0000	1.0000
1.17	1.0000	1.0000	1.0000	1.0000	1.0000
1.33	1.0000	1.0000	1.0000	1.0000	1.0000
1.50	1.0000	1.0000	1.0000	1.0000	1.0000
1.67	1.0000	1.0000	1.0000	1.0000	1.0000
1.83	1.3165	1.3440	1.3502	1.2831	1.3535
2.00	1.2990	1.3215	1.3329	1.2722	1.3335
2.17	1.2809	1.2984	1.3150	1.2611	1.3150
2.33	1.2626	1.2753	1.2967	1.2499	1.2967
2.50	1.2441	1.2517	1.2781	1.2384	1.2781
2.67	1.2253	1.2291	1.2592	1.2268	1.2601
2.83	1.2064	1.2101	1.2403	1.2151	1.2422
3.00	1.1911	1.1952	1.2228	1.2039	1.2251
3.17	1.1817	1.1861	1.2105	1.1974	1.2130
3.33	1.1784	1.1829	1.2057	1.1974	1.2085
3.50	1.1768	1.1805	1.2039	1.1977	1.2072
3.67	1.1750	1.1772	1.2024	1.1988	1.2066
3.83	1.1725	1.1735	1.2001	1.2027	1.2069
4.00	1.1694	1.1693	1.1973	1.2072	1.2079
4.17	1.1657	1.1643	1.1938	1.2109	1.2110
4.33	1.1619	1.1586	1.1896	1.2136	1.2137
4.50	1.1591	1.1538	1.1846	1.2154	1.2154
4.67	1.1571	1.1498	1.1790	1.2166	1.2166
4.83	1.1548	1.1454	1.1726	1.2177	1.2177
5.00	1.1520	1.1404	1.1654	1.2178	1.2178
5.17	1.1483	1.1349	1.1576	1.2162	1.2162
5.33	1.1436	1.1290	1.1492	1.2130	1.2130
5.50	1.1421	1.1215	1.1395	1.2113	1.2113
5.67	1.1423	1.1153	1.1315	1.2108	1.2108
5.83	1.1471	1.1169	1.1324	1.2149	1.2149
6.00	1.1570	1.1211	1.1385	1.2244	1.2244
6.17	1.1698	1.1286	1.1528	1.2369	1.2369
6.33	1.1807	1.1349	1.1652	1.2473	1.2473

Table A.1 (continued)
W(z) versus Core Height
(Top and Bottom 15% Excluded)

Height (feet)	150 MWD/MTU	4000 MWD/MTU	10000 MWD/MTU	18000 MWD/MTU	Bounding W(z)
6.50	1.1916	1.1439	1.1762	1.2572	1.2572
6.67	1.2031	1.1560	1.1864	1.2669	1.2669
6.83	1.2141	1.1680	1.1951	1.2744	1.2744
7.00	1.2239	1.1801	1.2025	1.2800	1.2800
7.17	1.2324	1.1925	1.2082	1.2837	1.2837
7.33	1.2394	1.2046	1.2119	1.2853	1.2853
7.50	1.2448	1.2159	1.2158	1.2849	1.2849
7.67	1.2485	1.2260	1.2201	1.2824	1.2824
7.83	1.2504	1.2348	1.2229	1.2780	1.2780
8.00	1.2501	1.2424	1.2258	1.2709	1.2709
8.17	1.2481	1.2486	1.2302	1.2623	1.2706
8.33	1.2466	1.2536	1.2352	1.2553	1.2709
8.50	1.2466	1.2563	1.2385	1.2492	1.2717
8.67	1.2472	1.2582	1.2414	1.2465	1.2726
8.83	1.2478	1.2639	1.2498	1.2508	1.2746
9.00	1.2508	1.2720	1.2624	1.2584	1.2790
9.17	1.2614	1.2816	1.2762	1.2655	1.2893
9.33	1.2750	1.2945	1.2893	1.2724	1.3028
9.50	1.2883	1.3177	1.2996	1.2780	1.3216
9.67	1.3003	1.3516	1.3066	1.2820	1.3524
9.83	1.3124	1.3867	1.3125	1.2866	1.3867
10.00	1.3248	1.4220	1.3175	1.2922	1.4220
10.17	1.3392	1.4558	1.3218	1.2979	1.4558
10.33	1.0000	1.0000	1.0000	1.0000	1.0000
10.50	1.0000	1.0000	1.0000	1.0000	1.0000
10.67	1.0000	1.0000	1.0000	1.0000	1.0000
10.83	1.0000	1.0000	1.0000	1.0000	1.0000
11.00	1.0000	1.0000	1.0000	1.0000	1.0000
11.17	1.0000	1.0000	1.0000	1.0000	1.0000
11.33	1.0000	1.0000	1.0000	1.0000	1.0000
11.50	1.0000	1.0000	1.0000	1.0000	1.0000
11.67	1.0000	1.0000	1.0000	1.0000	1.0000
11.83	1.0000	1.0000	1.0000	1.0000	1.0000
12.00 (top)	1.0000	1.0000	1.0000	1.0000	1.0000

Table A.2

F_Q Penalty Factors as a Function of Cycle Burnup

Cycle 15 Burnup	$F_Q^W(z)$ Penalty Factor (%)
1351	3.06
1523	3.43
1694	3.01
1866	2.55
2038	2.10

Note: All cycle burnups not in the range of the above table shall use a 2.0% penalty factor for compliance with Surveillance Requirement 3.2.1.2.

For values of burnup between two of those listed in the first column, the greater of the two corresponding penalty factors shall be used for compliance with Surveillance Requirement 3.2.1.2.

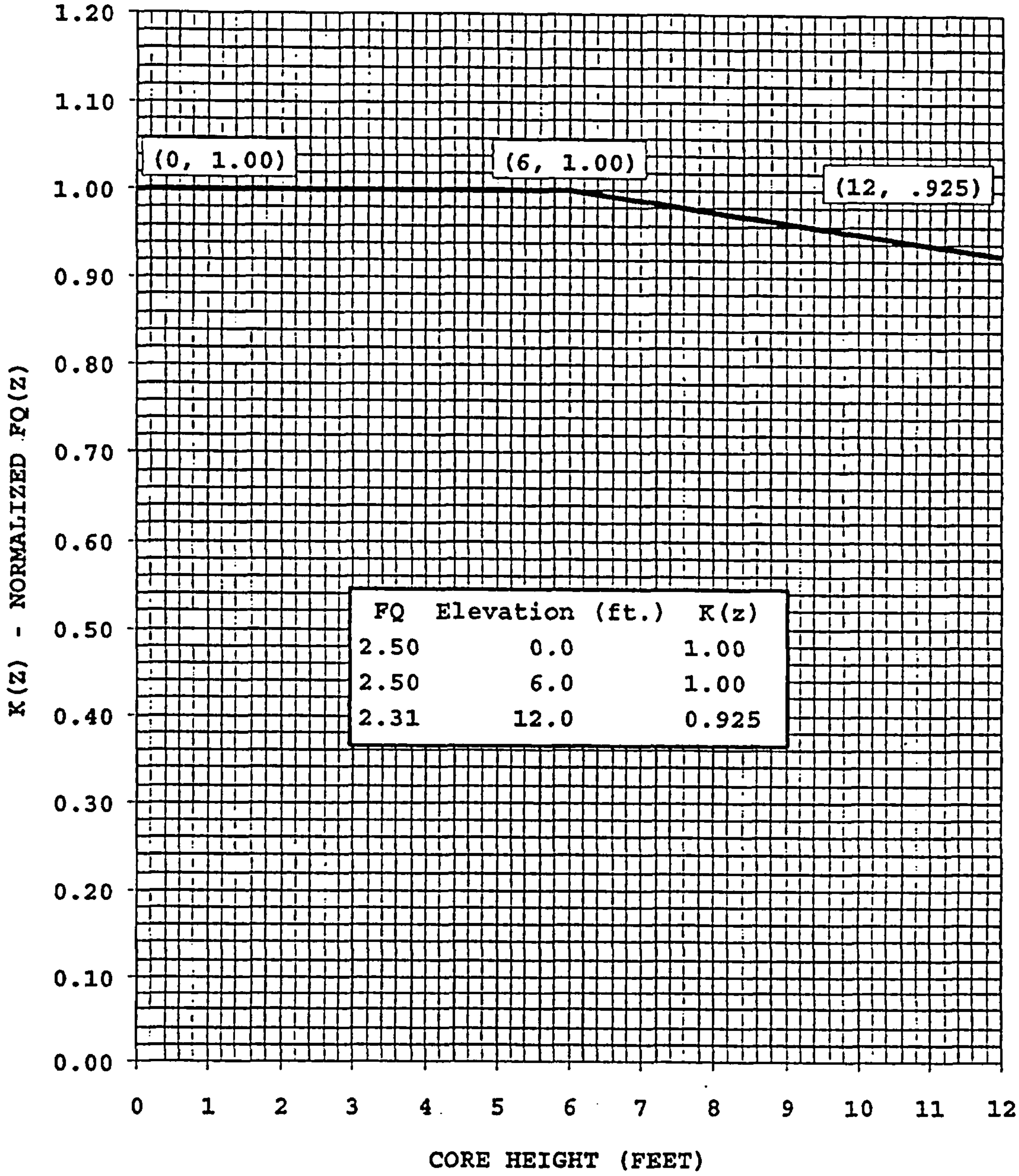

Figure 3

Callaway Cycle 15
K(z) - Normalized $F_Q(z)$
as a Function of Core Height

2.6 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^{N}$
(Specification 3.2.2)

$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} [1 + PF_{\Delta H}(1-P)]$

where: $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

2.6.1 $F_{\Delta H}^{RTP} = 1.59$

2.6.2 $PF_{\Delta H} = 0.3$

2.7 Axial Flux Difference
(Specification 3.2.3)

The Axial Flux Difference (AFD) Limits are provided in Figure 4.

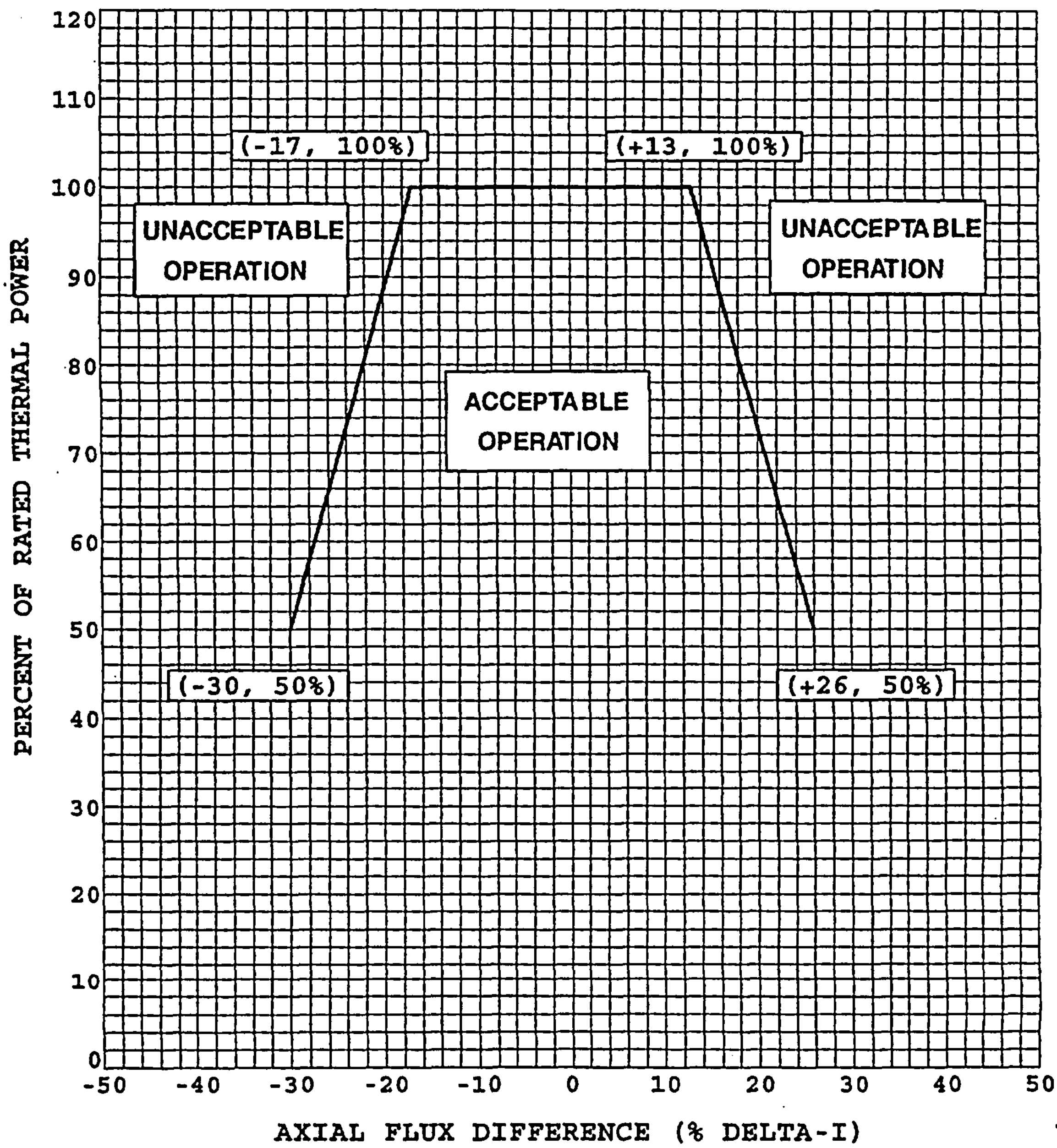

Figure 4 *

Callaway Cycle 15
Axial Flux Difference Limits as a Function of Rated Thermal Power for RAOC

* More restrictive AFD limits exist for Cycle 15 operation. Refer to Curve Book Figure 1-1

2.8 <u>Reactor Core Safety Limits</u>
(Safety Limit 2.1.1)

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits in Figure 5.

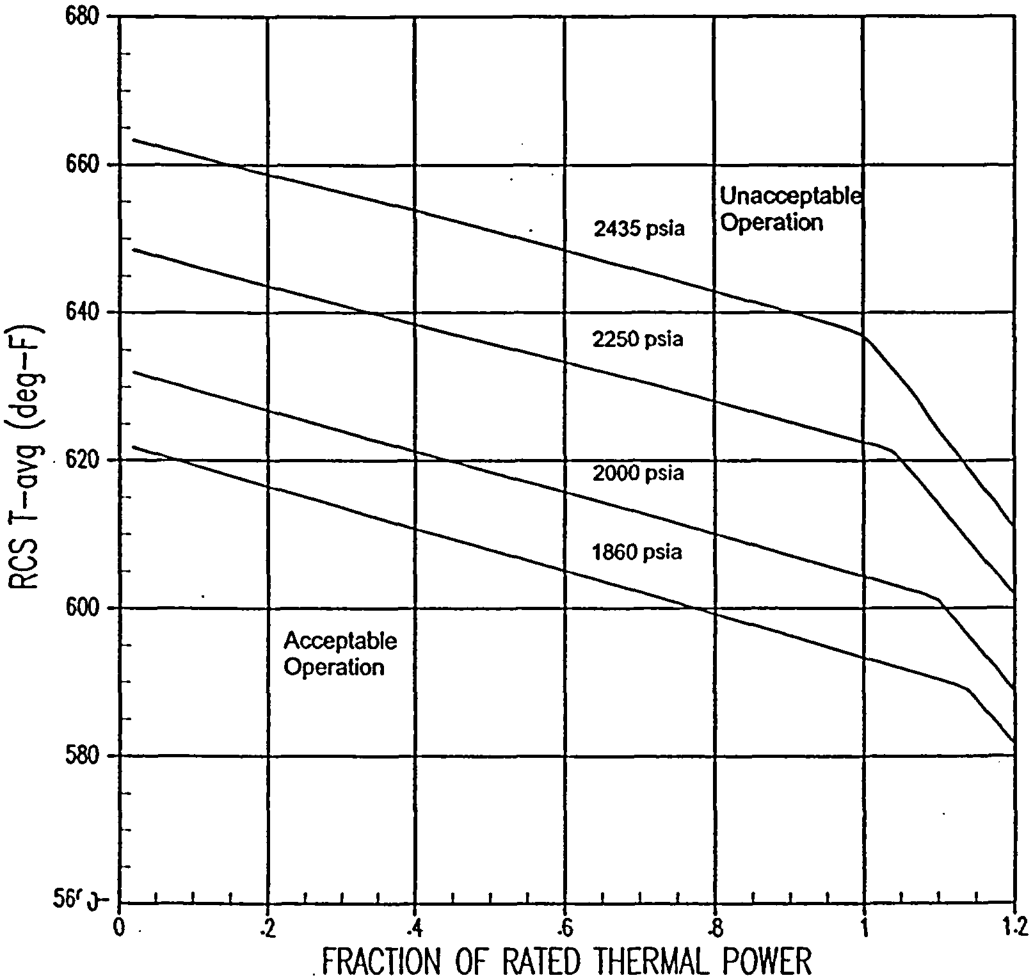

Figure 5

~~Figure 2.1.1-1 (page 1 of 1)~~
Reactor Core Safety Limits

2.9 Reactor Trip System Overtemperature ΔT Setpoint Parameter Values (Specification 3.3.1)

Parameter	Value
Overtemperature ΔT reactor trip setpoint	$K_1 = 1.1950$
Overtemperature ΔT reactor trip setpoint T_{avg} coefficient	$K_2 = 0.0251/°F$
Overtemperature ΔT reactor trip setpoint pressure coefficient	$K_3 = 0.00116/psig$
Nominal T_{avg} at RTP	$T' \leq 585.3$ °F
Nominal RCS operating pressure	$P' = 2235$ psig
Measured RCS ΔT lead/lag time constants	$\tau_1 \geq 8$ sec $\tau_2 \leq 3$ sec
Measured RCS ΔT lag time constant	$\tau_3 = 0$ sec
Measured RCS average temperature lead/lag time constants	$\tau_4 \geq 28$ sec $\tau_5 \leq 4$ sec
Measured RCS average temperature lag time constant	$\tau_6 = 0$ sec

$f_1(\Delta I) = -0.0325\ \{21\% + (q_t - q_b)\}$	when $(q_t - q_b) < -21\%$ RTP
0	when -21% RTP $\leq (q_t - q_b) \leq 8\%$ RTP
$0.02973\ \{(q_t - q_b) - 8\%\}$	when $(q_t - q_b) > 8\%$ RTP

Where, q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

2.10 Reactor Trip System Overpower ΔT Setpoint Parameter Values
(Specification 3.3.1)

Parameter	Value
Overpower ΔT reactor trip setpoint	$K_4 = 1.1073$
Overpower ΔT reactor trip setpoint T_{avg} rate/lag coefficient	$K_5 = 0.02/°F$ for increasing T_{avg} $= 0/°F$ for decreasing T_{avg}
Overpower ΔT reactor trip setpoint T_{avg} heatup coefficient	$K_6 = 0.0015/°F$ for $T > T''$ $= 0/°F$ for $T \leq T''$
Nominal T_{avg} at RTP	$T'' \leq 585.3°F$
Measured RCS ΔT lead/lag time constants	$\tau_1 \geq 8$ sec $\tau_2 \leq 3$ sec
Measured RCS ΔT lag time constant	$\tau_3 = 0$ sec
Measured RCS average temperature lag time constant	$\tau_6 = 0$ sec
Measured RCS average temperature rate/lag time constant	$\tau_7 \geq 10$ sec

$f_2(\Delta I) = 0$ for all ΔI.

2.11 RCS Pressure and Temperature Departure from Nucleate Boiling (DNB) Limits
(Specification 3.4.1)

Parameter	Indicated Value
Pressurizer pressure	$\geq$ 2223 psig
RCS average temperature	$\leq$ 590.1 °F

APPENDIX A

Approved Analytical Methods for Determining Core Operating Limits

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:

1. WCAP-9272-P-A. "Westinghouse Reload Safety Evaluation Methodology," July 1985.

   NRC letter dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report WCAP-9272(P)/9273(NP), "Westinghouse Reload Safety Evaluation Methodology"."

2. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

   NRC Safety Evaluation Report dated November 26, 1993, "Acceptance for Referencing of Revised Version of Licensing Topical Report WCAP-10216-P, Rev. 1, Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification" (TAC No. M88206).

3. WCAP-10266-P-A, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

   NRC letter dated November 13, 1986, "Acceptance for Referencing of Licensing Topical Report WCAP-10266 "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code.""

   WCAP-10266-P-A, Addendum 1, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 1: Power Shape Sensitivity Studies," December 1987.

   NRC letter dated September 15, 1987, "Acceptance for Referencing of Addendum 1 to WCAP-10266, BASH Power Shape Sensitivity Studies."

   WCAP-10266-P-A, Addendum 2, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 2: BASH Methodology Improvements and Reliability Enhancements," May 1988

   NRC letter dated January 20, 1988, "Acceptance for Referencing Topical Report Addendum 2 to WCAP-10266, Revision 2, "BASH Methodology Improvements and Reliability Enhancements."

4. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.

   NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610, 'VANTAGE+ Fuel Assembly Reference Core Report' (TAC NO. 77268)."

   NRC Safety Evaluation Report dated September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC NO. M86416)."

5. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

   NRC Safety Evaluation Report dated January 17, 1989, "Acceptance for Referencing of Licensing Topical Report WCAP-11397, "Revised Thermal Design Procedure."

6. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

   NRC letter dated January 19, 1999, "Acceptance for Referencing of Licensing Topical Report WCAP-14565, 'VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis' (TAC No. M98666)."

7. WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," August 1988.

   NRC letter dated May 9, 1988, "Westinghouse Topical Report WCAP-10851, 'Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations.'"

8. WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000.

   NRC letter dated April 24, 2000, "Safety Evaluation Related to Topical Report WCAP-15063, Revision 1, 'Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)' (TAC NO. MA2086)."

9. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

   NRC Safety Evaluation Report dated April 17, 1986, "Acceptance for Referencing of Licensing Topical Report WCAP-8745(P)/8746(NP), 'Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions.'"

10. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.

    NRC letter dated June 23, 1986, "Acceptance for Referencing of Topical Report WCAP 10965-P and WCAP 10966-NP."

11. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

    NRC Safety Evaluation Report dated May 17, 1988, "Acceptance for Referencing of Westinghouse Topical Report WCAP-11596 - Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

12. WCAP-13524-P-A, "APOLLO: A One Dimensional Neutron Diffusion Theory Program," Revision 1-A, September 1997.

    NRC letter dated June 9, 1997, "Acceptance for Referencing of Licensing Topical Reports WCAP-13524 and WCAP-13524, Revision 1, 'APOLLO – A One-Dimensional Neutron Diffusion Theory Program.'"