Westinghouse Non-Proprietary Class 3

WCAP-14483-A

Generic Methodology for Expanded Core Operating Limits Report

Westinghouse Energy Systems

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14483-A

GENERIC METHODOLOGY

FOR

EXPANDED CORE OPERATING LIMITS REPORT

MUHP-1008

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November 1995

Approved January 19, 1999

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 19, 1999

Mr. Andrew Drake
Westinghouse Owners Group
Westinghouse Electric Corporation
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, PA 15320-0355

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT WCAP-14483, "GENERIC METHODOLOGY FOR EXPANDED CORE OPERATING LIMITS REPORT" (TAC NO. M94338)

Dear Mr. Drake:

The staff has reviewed the subject report submitted by the Westinghouse Owners Group (WOG) by letter of December 1, 1995, and supplemented by letter of November 25, 1998. The report is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the report. The WOG should submit a technical specification task force (TSTF) change to the standard technical specifications to implement any of the approved changes described in the safety evaluation.

The staff will not repeat its review of the matters described in WCAP-14483 when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in WCAP-14483 and in the response to the staff request for additional information. In accordance with procedures established in NUREG-0390, the NRC requests that the WOG publish accepted versions of the submittal within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract and a "-A" (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion that the submittal is acceptable is invalidated, the WOG and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Thomas H. Essig

Thomas H. Essig, Acting Chief
Generic Issues and Environmental Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation for WCAP-14483

RECEIVED
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ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO TOPICAL REPORT WCAP-14483
GENERIC METHODOLOGY FOR EXPANDED CORE OPERATING LIMITS REPORT

1. INTRODUCTION

In a letter of March 8, 1996, from T. V. Greene to the U. S. Nuclear Regulatory Commission (NRC), the Westinghouse Owners Group (WOG) submitted Topical Report WCAP-14483 for NRC review (Ref. 1). The purpose of the topical report is to provide justification to support the technical specification (TS) changes required to expand current Core Operating Limits Reports (COLRs) associated with Westinghouse plants. Specifically, NRC approval of the report would allow the departure-from-nucleate-boiling (DNB) parameters of reactor coolant system (RCS) average temperature (T-avg), RCS flow rate, and pressurizer pressure, as well as the overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) trip setpoint parameter values to be relocated to the COLR. The current reactor core safety limit figure would be relocated to the COLR and replaced with the fuel DNB ratio (DNBR) limit and the fuel centerline temperature limit (Ref. 2).

2. SUMMARY OF TOPICAL REPORT

Section 1.0 gives a general introduction which includes the background, purpose, and contents of the report. The bases for the DNB and the OTΔT and OPΔT parameter values are stated in Sections 2.0 and 3.0, respectively, as well as the bases and benefits for relocating these values to the COLR. The basis for the reactor core safety limits figure and the basis and benefits of replacing the figure are given in Section 4.0. Conclusions are given in Section 5.0 and references in Section 6.0. Sample revised WOG improved TS markups based on NUREG-1431, Revision 1, are given in Appendix A and sample COLR revisions are given in Appendix B.

3. TECHNICAL EVALUATION OF REPORT

NRC Generic Letter 88-16 (Ref. 3) allows licensees to remove cycle-dependent variables from TS provided that the values of these variables are included in a COLR and are determined with NRC-approved methodology which is referenced in the TS. These variables are moved from TS to the COLR to avoid the need for frequent revision of TS to change the value of those operating limits which cannot be specified to reasonably bound several operating cycles without significant loss of operating flexibility.

A number of Westinghouse-designed plants have been permitted to remove the cycle-specific values in certain limiting conditions for operation (LCOs) from plant TS and to place them in a COLR. Safety limits, however, may not be placed in the COLR. Several cycle-specific TS parameters of Westinghouse plants have been approved by the NRC for inclusion in COLRs. These include the following:

(1) moderator temperature coefficient (MTC)

(2) shutdown bank insertion limits

(3) control bank insertion limits

(4) axial flux difference limits

(5) nuclear heat flux hot channel factor limit, F_Q

(6) nuclear enthalpy rise hot channel factor limit, $F^N_{\Delta H}$

(7) refueling boron concentration

(8) shutdown margin (SDM)

The NRC has also previously extended this philosophy to the cycle-dependent OTΔT and OPΔT setpoint parameters and function modifiers for the Catawba (Ref. 4), McGuire (Ref. 5), and Seabrook (Ref. 6) Nuclear Stations. This allows these setpoints to be based on cycle-specific core design parameters, which are verified on a cycle-specific basis, thereby avoiding the necessity of overly conservative TS limits. The applicable NRC-approved setpoint methodology, WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986, (or other applicable setpoint methodology) is referenced in the administrative reporting requirements section of the TS. Therefore, the WOG proposal to relocate the OTΔT and OPΔT setpoint parameter values to the COLR is acceptable.

The TS limits on the DNB parameters assure that pressurizer pressure, RCS flow, and the RCS T-avg will be maintained within the limits of steady-state operation assumed in the accident analyses. These limits must be consistent with the initial full power conditions considered in the FSAR safety analysis for normal operation and anticipated operational occurrences (AOOs) in which precluding DNB is the primary criterion. The DNB parameter limits are also based on initial conditions assumed for accidents in which precluding DNB is not a criterion.

A number of WOG licensees have implemented T-Hot Reduction and steam generator tube plugging programs. In these cases, additional margin has been allocated to support the TS and to minimize any licensing impacts associated with cycle-to-cycle changes in RCS T-avg and RCS flow rate. In addition, some licensees have performed safety analyses which support plant operation at different nominal operating pressures. In these cases, additional margin must be allocated for the pressurizer pressure TS to reflect the most limiting value assumed in the safety analyses to avoid cycle-specific TS changes. Therefore, although these plants may operate with a full power T-avg that is lower than the licensed upper T-avg limit, with higher RCS flow rates than assumed in the tube plugging analysis (due to actual lower steam generator tube plugging levels), or with lower operating pressures, the reactor protection system setpoints must be based on the limiting TS values since the safety analyses were based on these conservative TS values. By relocating these DNB TS parameters to the COLR, the COLR values would reflect the cycle-specific operating conditions and allow reactor trip setpoints to be consistent with actual operating conditions, thereby avoiding the necessity of overly conservative TS limits.

Although some plants operate with lower steam generator tube plugging levels and thus higher RCS flow rates than those assumed in the safety analyses, a change in RCS flow is an indication of a physical change to the plant which should be reviewed by the NRC staff. Because of this, the staff recommended that if RCS flow rate were to be relocated to the COLR, the minimum limit for RCS total flow based on a staff approved analysis (e. g., maximum tube plugging) should be retained in the TS to assure that a lower flow rate than reviewed by the staff would not be used. The WOG concurred with this recommendation and modified the proposed TS accordingly (Ref. 2).

The staff concludes that relocation of the RCS DNB limits to the COLR is acceptable. The NRC-approved methodology used to derive the parameters in the figure is contained in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985, (or other applicable approved reload methodology), and will be referenced in the Reporting Requirements section of the TS.

The current TS figure (2.1.1-1) presents core limits on RCS temperature conditions (T-avg) as a function of pressurizer pressure and fractional rated thermal power. This figure was originally included in the Westinghouse TS to satisfy the requirements of 10CFR50.36 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." However, the figure is not a complete representation of reactor core safety limits but is intended to provide the relationship between the process variables that are available to the operator (i.e., T-avg, pressurizer pressure, and thermal power) and the DNB design basis safety limit.

To ensure that the requirements of 10CFR50.36 are met, i.e., limits upon important process variables, the WOG has proposed to retain the requirement for a Reactor Core Limits figure in the Safety Limits TS, but relocate the actual figure to the COLR and replace it with the DNB design basis limit and the fuel centerline melt limit (Ref. 2). Both of these limits are criteria that must be satisfied for normal operation and for AOOs to prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the RCS and are, therefore, the true safety limits. The reactor protection system (RPS) and the Reactor Core Limits figure would then be used to determine whether the actual DNB and fuel centerline melt safety limits were violated should an event occur that could potentially challenge them. Appropriate functioning of the RPS and the steam generator safety valves ensures that for variations in the thermal power, RCS pressure, RCS average temperature, RCS flow rate, and ΔI(percent power in top half of core minus percent power in bottom half of core), the reactor core safety limits will be satisfied during steady-state operation, normal operational transients, and AOOs. Therefore, in the event of an AOO, verification that the RPS and the main steam system safety valves are functioning as designed will ensure that all safety limits are met. In the event that the RPS is not functioning as designed, an evaluation of any transient condition would be required to determine whether or not the safety limits have been violated.

In addition, since the Reactor Core Limits figure is based on the nuclear enthalpy rise hot channel factor limit, $F^N_{\Delta H}$, and the RCS total flow rate, both of which may be in the COLR, relocation of the figure to the COLR would eliminate the need for a license amendment if cycle-dependent changes to these parameters were to exist.

The staff concludes that the Reactor Core Limits figure may be relocated to the COLR and replaced with the DNB design basis limit and the fuel centerline melt limit. The NRC-approved methodology used to derive the parameters in the figure is contained in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985, (or other applicable approved reload methodology), and will be referenced in the Reporting Requirements section of the TS.

4. CONCLUSION

The staff has reviewed the request by the WOG to implement the following TS changes for Westinghouse plants:

1. Revise TS 3.4.1 of NUREG-1431, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, to relocate the pressurizer pressure, RCS average temperature (T-avg), and RCS total flow rate values to the COLR. The minimum limit for total flow based on that used in the reference safety analysis will be retained in the TS.

2. Revise TS Table 3.3.1-1 of NUREG-1431, Reactor Trip System Instrumentation, to relocate the overtemperature ΔT and overpower ΔT (K) constant values and dynamic compensation (τ) values, and the breakpoint and slope values for the $f(\Delta I)$ penalty function(s) to the COLR.

3. Revise TS 2.1 Safety Limits of NUREG-1431, and the associated bases to relocate Figure 2.1.1-1 to the COLR and replace it with more specific requirements regarding the safety limits (i.e., the fuel DNB design basis and the fuel centerline melt design basis). The NRC-approved methodology used to derive the parameters in the figure will be referenced in the Reporting Requirements section of the TS.

Based on the above safety evaluation, we find the requested TS changes acceptable.

5. REFERENCES

1. Letter, T. V. Greene (WOG) to Document Control Desk (NRC), Westinghouse Owners Group - Review of WCAP-14483, "Generic Methodology for Expanded Core Operating Limits Report," March 8, 1996.

2. Letter, L. F. Liberatori Jr. (WOG) to Document Control Desk (NRC), Westinghouse Owners Group - Response to NRC Request for Additional Information on WCAP-14483, "Generic Methodology for Expanded Core Operating Limits Report," (Non-Proprietary), (MUHP1009), November 25, 1998.

3. NRC Generic Letter 88-16, "Guidance for Technical Specification Changes for Cycle-Specific Parameter Limits," October 4, 1988.

4. Letter, R. E. Martin (NRC) to D. L. Rehn (Duke Power), Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473), March 25, 1994.

5. Letter, V. Nerses (NRC) to T. C. McMeekin (Duke Power), Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2 (TAC Nos. M85474 and M85475), May 31, 1994.

6. Letter, A. W. DeAgazio (NRC) to T. C. Feigenbaum (North Atlantic Energy Service Corp.), Amendment No. 33 to Facility Operating License NPF-86: Wide-Band Operation and Core Enhancements - License Amendment Request 93-18 (TAC M87847), November 23, 1994.

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1.0 Introduction

1.1 Background

In 1988, the Nuclear Regulatory Commission (NRC) issued Generic Letter 88-16, entitled "Removal of Cycle-Specific Parameter Limits from Technical Specifications (TS)." The purpose of this Generic Letter was to provide 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 recommended 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 this information even though NRC approval of these limits would not be required.

In response to this Generic Letter, the Westinghouse Owners Group (WOG) authorized the development of a generic COLR License Amendment Request (LAR) package for Westinghouse plants. The generic COLR LAR package included provisions for the TS limits presented in Table 1. The generic package, along with background and supplemental information, was provided to the WOG members. This COLR process has been implemented by a number of WOG licensees and has greatly improved the reload process, giving plants enhanced core design flexibility without the need for cycle-specific license amendments. This has also provided licensees with the flexibility to address cycle-specific issues without the need for cycle-specific licensing submittals.

The concept of the COLR process described in Generic Letter 88-16 has also been utilized for relocating the heatup and cooldown figures and Cold Overpressurization Mitigation System (COMS) setpoints to a Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) (Reference 1), which is similar to what was done for the core related TS parameter limits presented in Table 1.

Table 1

Technical Specification Limits Relocated to the COLR

Technical Specification Item	NUREG-1431
Moderator Temperature Coefficient	3.1.4
Shutdown Bank Insertion Limit	3.1.6
Control Bank Insertion Limits	3.1.7
Axial Flux Difference Limits	3.2.3
Heat Flux Hot Channel Factor	3.2.1
Nuclear Enthalpy Rise Hot Channel Factor	3.2.2
Boron Concentration	3.9.1

1.2 Purpose

The purpose of this report is to provide justification to support the TS changes required to expand current COLRs to include cycle-specific RCS related TS parameter limits. This would allow licensees the flexibility to enhance plant operating margin and/or core design margins without the need for cycle-specific LARs, which is similar to what is currently done for the core related parameter limits presented in Table 1. The TS changes proposed for this program include the following.

1. Revise Technical Specification 3.4.1 of NUREG-1431, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, to relocate the Pressurizer pressure, RCS average temperature (T-avg), and RCS total flow rate values to the COLR.

2. Revise Technical Specification Table 3.3.1-1 of NUREG-1431, Reactor Trip System Instrumentation, to relocate the Overtemperature (OTDT) and Overpower (OPDT) K constant values and dynamic compensation tau (τ) values, and the breakpoint and slope values for the f(ΔI) penalty function(s) to the COLR.

3. Revise Technical Specification 2.1 Safety Limits of NUREG-1431, and the associated bases to replace Figure 2.1.1-1, Reactor Core Safety Limits (RCSLs) with more specific requirements regarding the safety limits (i.e., the fuel DNB design basis and the fuel centerline melting design basis).

It should be noted that the NRC has recently approved COLR additions, submitted by Duke Power, for item 2 for the Catawba and McGuire units (see References 2 and 3), to allow for cycle-specific margin utilization. The Duke Power submittal was based on NRC-approved Duke Power, B&W, and Westinghouse analytical methods. The NRC has also approved relocation of the RCS Total Flow Rate of item 1 to the COLR for the Prairie Island units (Reference 4).

In addition to providing justification for the above proposed TS changes and moving of the TS parameters to the COLR, the report provides a generic change package containing markups of the affected TS and Bases for NUREG-1431, as well as sample COLR revisions, following the guidelines provided in NRC Generic Letter 88-16.

It should be noted that this report is also applicable to those plants with the Overtemperature and Overpower N-16 thermal protection functions.

1.3 Contents

This report contains the justification, based on NRC-approved methodologies, to license the following TS changes:

1. Relocate the DNB parameters of RCS T-avg, RCS flow rate and Pressurizer pressure to the COLR

2. Relocate the OTDT and OPDT setpoint parameter values to the COLR

3. Replacement of the Reactor Core Safety Limit Figure with the fuel DNB and the fuel centerline melting requirements and design bases

This report also includes markups of the affected Technical Specifications and Bases for NUREG-1431, as well as sample COLR additions. While this report presents markups for NUREG-1431 TS, the information and markups presented within are applicable in principle and can be applied, with appropriate modifications, to plants with "Custom" and NUREG-0452 format TS.

2.0 Relocation of the DNB Parameter Limit Values to COLR

2.1 Basis for the DNB Parameter Limits

The bases for the DNB Parameter TS limits are strictly related to the assumptions on plant initial conditions used in the accident analyses presented in the Final Safety Analysis Report (FSAR). The limits of the DNB Parameter TS assure that the parameters of pressurizer pressure, reactor coolant system flow, and the reactor coolant system T-avg are 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 accident analysis presented in the FSAR. For those events for which precluding DNB is the primary criterion (i.e., Condition I and II events) the safety analyses have demonstrated that the DNB design basis is satisfied, assuming that the plant is operating in compliance with the TS requirements, and in particular with the DNB parameter limits, prior to the initiation of the event. In addition, the DNB parameter limits are also based on the initial conditions assumed for events for which precluding DNB is not a criterion, that is, Condition III and IV events.

It is important to note that the limits on these process variables should not be based on the reactor core safety limits (Figure 2.1.1-1) as this figure is intended to define acceptable conditions for Condition I and II class events. This figure and its limitations are discussed further in section 4.0.

Given that the DNB parameter TS presents limits on important process variables, which ensure that the DNB design basis and other safety criteria are satisfied, continuous operation at less limiting conditions would generate margin to these safety criteria. This is discussed further in the following sections.

2.2 Basis for Moving the DNB Parameter Limit Values

A significant number of WOG licensees have performed and implemented T-Hot Reduction and Steam Generator Tube Plugging programs which allow for a wide range of operating configurations in RCS T-avg and RCS flow rate space. Analyses supporting T-Hot Reduction windows on the order of 15 to 20°F and steam generator tube plugging levels on the order of 25 to 30%, have been performed for a number of licensees. In addition, some licensees have performed and implemented analyses which support operation at different nominal operating pressures. The analyses performed to support these programs allow licensees the flexibility to define nominal operating conditions on a cycle-specific basis without the need to reanalyze any of the safety analyses or change any of the TS limits. This is accomplished by conservatively bounding the full range of operating conditions being considered. With the implementation of these programs, a plant can operate, for instance, at the lowest possible full power T-avg while maintaining full rated power thereby allowing the plant to enhance steam generator performance.

The limits of the conservative assumptions made are reflected in the TS limits and minimize any cycle-specific licensing impacts associated with LAR submittals, since these can be very costly and may adversely impact restart schedules. The disadvantage of this approach is that margin allocated to support the conservative TS limits cannot be easily utilized on a cycle-specific basis without an LAR submittal.

In the case of plants that have implemented a T-Hot Reduction program, a full power T-avg is selected for a given cycle which is typically less than the licensed (TS) upper T-avg limit in order to enhance steam generator performance. This cycle-specific full power T-avg is a key reload design basis parameter, as noted in the Westinghouse Reload Methodology (see Reference 5). For reload methodologies, such as the Westinghouse Reload Methodology, the analysis assumptions are verified as part of the reload process to ensure that the reference safety analyses remain valid on a cycle-specific basis. The full power T-avg is a key core reload design analysis parameter primarily because the moderator temperature coefficient of reactivity (and other reactivity parameters to a lesser extent) is temperature dependent. The confirmation of the different reactivity related parameters assumed in the safety analyses is performed using the cycle-specific nominal full power operating T-avg, as described in Reference 5. Thus, the cycle-specific full power T-avg is directly tied to the core reload and the core reload process as it affects the safety analyses.

In addition, the nominal full power T-avg also forms the basis for the control system setpoints (e.g., rod control system programmed T-avg, steam dump system, etc.) and the reactor protection system setpoints, specifically the OTDT and OPDT reactor trip setpoint T' and T" values. At the beginning of a cycle, these different systems are scaled to be consistent with the cycle-specific full power T-avg. However, the safety analyses cannot credit a reduced cycle-specific full power T-avg because the safety analyses must support the DNB parameter TS T-avg limit (which is based on the highest T-avg the analyses support). In such an instance, analytical and operating margin is essentially "lost" on a cycle-specific basis.

The same situation described above for the RCS T-avg also applies to the DNB Parameter TS RCS total flow rate. Plants are analyzed with conservatively high steam generator tube plugging levels, relative to actual plant steam generator tube plugging levels. The effect of these analyzed steam generator tube plugging levels is reflected in the DNB Parameter TS RCS total flow rate. Many plants operate with much lower steam generator tube plugging levels (and thus higher RCS total flow rates) compared to what is assumed in the safety analyses. This results in RCS flow margin which could be utilized on a cycle-specific basis rather than being allocated to support the existing TS. As was the case for the RCS T-avg, the control systems and protection systems are scaled to be consistent with the actual cycle-specific RCS total flow rate. For instance, it is recommended that the RCS low flow rate reactor trip is set to be consistent with the actual RCS flow rate measured in each corresponding loop (presuming the measured loop flow rate is greater than the minimum required flow rate). In addition, the reference ΔT_0 for the OTDT and OPDT setpoints in each loop reflects the actual RCS flow rate in each corresponding loop via use of the loop specific indicated ΔT at full power conditions. However, the safety analyses cannot credit the actual RCS total flow rate because the analyses must support the DNB Parameter TS RCS total flow rate limit.

Finally, some licensees have safety analyses which support plant operation at different nominal operating pressures. To support the different possible operating pressures, the safety analyses have margin allocated to support the DNB Parameter Pressurizer pressure limit that could otherwise be used on a cycle-specific basis. The OTDT reactor trip setpoint as well as the pressurizer pressure control system is typically set to be consistent with the nominal operating pressure that is selected for a given cycle. However, the DNB Parameter TS Pressurizer pressure limit must reflect the most limiting value assumed in the safety analyses to avoid cycle-specific TS changes.

In conclusion, the planned cycle-specific operating configuration, that is, the RCS T-avg, RCS total flow rate and pressurizer pressure, is assumed in the core reload design process (described in Reference 5). This reload design process demonstrates that the reload related parameters assumed in the safety analyses are valid for the cycle in question. This ensures that the safety analyses remain bounding and the conclusions of the FSAR remain valid. To minimize any licensing impacts associated with cycle-to-cycle changes in RCS T-avg and RCS flow rate, significant margin is allocated to support the current TS for those plants that have been analyzed for a T-Hot window and/or increased levels of steam generator tube plugging. In addition, some plants have margin allocated for the pressurizer pressure TS because the analyses support more than one nominal operating pressure. To better utilize the margin currently allocated to support the existing limit values for the RCS T-avg, RCS total flow rate, and Pressurizer pressure of the DNB Parameter TS, it is proposed that these parameters be relocated to the COLR, consistent with the recommended guidance presented in the NRC Generic Letter 88-16.

It should be noted that the process of margin recovery associated with core design related COLR parameters is presently ongoing for many licensees. In the same way, the values for the DNB parameters of RCS T-avg, RCS total flow rate and pressurizer pressure could be relocated to the COLR to ensure that available margins are not unnecessarily allocated and "lost" on a cycle-specific basis just to support overly conservative TS limits. As noted previously, the proposed TS/COLR changes would reflect the safety analyses assumptions, consistent with what is currently performed in the core reload design process, which is; to specifically design and analyze the cores consistent with the planned cycle-specific operating configuration. By using the cycle-specific RCS parameters, the TS/COLR and the safety analyses would more closely reflect the cycle-specific conditions that the plant control and protection systems are set to for a given cycle. All of this could be accomplished with existing NRC-approved methods and without the need for major reanalysis.

In addition, should a licensee choose to perform cycle-specific analyses to optimize the use of margins, having the DNB parameter limits in the COLR would allow the changes to be made without TS changes. Changes to the DNB parameter limits are burdensome, time consuming, and are "trivial" reviews, since all the analyses would be performed in accordance with NRC approved methodologies, per the COLR TS.

The resulting TS and Bases changes and sample COLR additions are presented in Appendices A and B, respectively.

2.3 Benefits of Moving the DNB Parameter Limit Values

The benefit of relocating the DNB Parameter limit values to the COLR is that it would allow licensees the flexibility to utilize available margins to increase cycle operating margins and/or improve core reload designs. This margin could also be used to offset any DNB penalties, address licensing issues, etc., without the requirement of cycle-specific license amendments. For example, a plant operating with a reduced full power T-avg could use the margin gained to help justify an increase in the $F_{\Delta H}$ limit.

The relocation of these selected TS limit values to the COLR would result 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.

3.0 Relocation of the OTDT and OPDT Setpoint Parameter Values to the COLR

3.1 Basis for the OTDT/OPDT Setpoints

The basis for the OTDT and OPDT reactor trip functions is to ensure that during any Condition I or II transient, there is at least a 95% probability at a 95% confidence level that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature. To achieve this, a fuel centerline temperature limit has been established (Reference 6) based on the melting temperature for UO_2 of 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU of burnup (Reference 7). For design purposes, this fuel centerline temperature limit is significantly below the melting temperature to allow for fuel temperature calculation and other uncertainties. In addition, the departure from nucleate boiling (DNB) design basis is defined as the probability that DNB will not occur on the limiting fuel rod(s) is at least 95% at a 95% confidence level. 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.

The OPDT reactor trip function, in conjunction with the OTDT reactor trip function, ensures operation within the fuel temperature design basis. With Westinghouse PWRs, this is accomplished through the OPDT trip function by correlating the core thermal power with the temperature difference across the vessel (ΔT). Since the thermal power is not precisely proportional to ΔT, because of the effects of changes in coolant density and heat capacity, a compensation term, which is a function of the vessel average temperature, is factored into the calculated overpower trip setpoint. A typical OPDT equation is presented below.

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{1}{(1+\tau_3)} \leq \Delta T_0 \{K_4 - K_5 (\frac{\tau_7 S}{(1 + \tau_7 S)}) \frac{1}{(1+\tau_6)} T - K_6 [T \frac{1}{(1+\tau_6)} - T''] - f_2(\Delta I)\}$$

where:

ΔT	=	measured RCS vessel ΔT
ΔT_o	=	vessel ΔT preset to the indicated ΔT at rated thermal power at the reference T″ (°F)
K_4	=	a preset manually adjustable bias (fraction of full-power ΔT)
K_5	=	a constant that compensates for piping and thermal time delays (fraction of full-power ΔT/°F). This term is zero for a constant T-avg because it is preceded by a rate lag compensation term.
K_6	=	a constant that compensates for the effects of coolant density and heat capacity on the relationship between ΔT and thermal power (fraction of full-power ΔT/°F)

T	=	indicated average RCS temperature (°F)
T″	=	indicated average RCS temperature at full power used in the calibration of the ΔT instrumentation (°F)
$f_2(\Delta I)$	=	a penalty, if required, that varies as a function of ΔI to account for adverse axial power distributions (fraction of full-power ΔT)
τ s	=	dynamic compensation time constants (sec)
s	=	Laplace transform variable (1/sec)

As noted above, the setpoint is set to be consistent with the nominal full power operating conditions. If a plant is operating at a reduced T-avg, the T" reference temperature is set to be consistent with the reduced full power T-avg. Likewise, the ΔT_o is set to be consistent with the measured ΔT associated with the reduced full power T-avg.

The OTDT reactor trip function, in conjunction with the OPDT 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 and pressure as well as the core thermal power, the OTDT reactor trip function is correlated with the vessel ΔT, the RCS T-avg, and pressurizer pressure. A compensating term which is a function of ΔI is also factored into the OTDT setpoint to account for the effect of changes in the axial power shape. A typical OTDT equation is presented below.

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

where:

ΔT	=	measured RCS vessel ΔT
ΔT_o	=	indicated vessel ΔT at rated thermal power (°F)
K_1	=	a preset manually adjustable bias (fraction of full-power ΔT)
K_2	=	a constant based on the effect of temperature on the design limits (fraction of full-power ΔT/°F)
K_3	=	a constant based on the effect of pressure on the design limits (fraction of full-power ΔT/psi)
T	=	indicated average RCS temperature (°F)
T′	=	indicated average RCS temperature at full power (°F)
P	=	pressurizer pressure (psig)
P′	=	nominal RCS pressure (psig)

$f_1(\Delta I)$ = a penalty that varies as a function of ΔI to account for adverse axial power distributions (fraction of full-power ΔT)

τs = dynamic compensation time constants (sec)

s = Laplace transform variable (1/sec)

As noted above, the setpoint is scaled to be consistent with the full power operating condition. If a plant is operating at a reduced T-avg, the T' reference temperature is set to be consistent with the reduced full power T-avg. Likewise, the ΔT_0 is set to be consistent with the measured ΔT associated with the reduced full power T-avg.

For those plants with the Overtemperature and Overpower N-16 thermal protection functions, the same design basis applies. The OTN-16 and OPN-16 functions use the measured N-16 level, whereas the OTDT and OPDT trips utilize ΔT as an indication of power, In addition, the OTN-16 and OPN-16 functions use the inlet temperature rather than the RCS T-avg.

3.2 Basis for Moving the OTDT/OPDT Setpoint Parameter Values

The justification for moving the OTDT and OPDT setpoint parameter values (Ks, τs, T', T'', P', and f(ΔI) functions) to the COLR is based on several considerations. These considerations are based primarily on the fact that the OTDT and OPDT setpoints are based on several parameters which are considered to be important reload design parameters. This is discussed further below.

A. The design basis of the OTDT reactor trip setpoint presented above, in conjunction with the OPDT setpoint also presented above, is to ensure that on a 95/95 basis that DNB is precluded. The OTDT and OPDT setpoints are calculated using the Reactor Core Safety Limits (RCSLs) and the Axial Offset Limits, as described in Reference 6. The RCSLs present the locus of RCS T-inlet conditions at various pressures and power levels, for a specific RCS total flow rate and a limiting reference axial power shape, where the DNBR safety analysis limit is satisfied and where exit boiling is precluded. The DNB limits are calculated using the $F\Delta H$ which is the enthalpy rise in the hottest channel of the core relative to the enthalpy rise in the average channel of the core. The RCSLs are a key reload design input which is verified on a cycle-specific basis as part of the reload process, described in Reference 5. Changes in reload related parameters, such as the $F\Delta H$, can impact the RCSLs and thus the OTDT (and OPDT) reactor trip setpoints on a cycle specific basis.

B. The Axial Offset Limits are used to generate the f(ΔI) penalty function of the OTDT setpoint which reduces the setpoint for highly skewed axial power shapes to ensure that the DNB design basis is satisfied. The Axial Offset Limits are calculated based on the allowable peaking factors and the axial offset control strategy used for normal operation and are verified on a cycle-specific basis. The peaking factors and axial offset control strategy parameters are TS limits whose values have been relocated to the COLR for many licensees. A change in any of these limit values could result in a change to the Axial Offset Limits and thus a change to the OTDT reactor trip setpoint on a cycle-specific basis.

C. The OTDT and OPDT setpoints are included in the reload process defined in Reference 5 and can be used to ensure that fuel design criteria are satisfied. It is possible that the OTDT/OPDT setpoints may need to be revised on a cycle-specific basis to ensure that the fuel rod design criteria are satisfied. Thus, it is possible that cycle-specific reload designs could result in changes to the OTDT and/or OPDT reactor trip setpoints.

D. For plants that have been analyzed for a T-avg (T-Hot) window, one OTDT setpoint and one OPDT setpoint are calculated for the entire T-avg window. When a plant operates at a reduced full power T-avg, the T' of the OTDT reactor trip function and the T" of the OPDT reactor trip function are set to be consistent with the reduced full power T-avg value. This effectively generates significant DNB margin and/or operating margin because the DNB limits that the OTDT and OPDT setpoints protect are independent of the full power operating T-avg. Thus, as the full power operating T-avg is reduced, the margin from the OTDT setpoint to the DNB limits increases. With the OTDT and OPDT setpoint parameter values in the TS, licensees cannot take advantage of this margin to improve the setpoints and enhance plant operating margins without prior NRC approval.

Given the above, there is sufficient justification for moving the OTDT and OPDT setpoint parameter values to the COLR. This justification includes 1) the setpoints are based on core design parameters which are verified on a cycle-specific basis, 2) the setpoints can be used on a cycle-specific basis to verify fuel design criteria, and 3) the setpoints typically have significant amounts of margin built into them, especially for plants that have T-avg windows, which currently cannot be fully utilized.

3.3 Benefits of Moving the OTDT/OPDT Setpoint Parameter Values

The benefits of moving the OTDT and OPDT reactor trip setpoint parameter values to the COLR were noted in the previous discussions. That is, moving the OTDT and OPDT setpoint parameter values to the COLR would minimize the chance that a reload related parameter change would necessitate a TS change. In addition, margin that is currently tied up in the setpoints for plants with T-avg windows could be utilized to provide enhanced setpoints. This is explained below.

As noted previously, plants that have been analyzed for a T-avg window have only one OTDT and one OPDT setpoint. When a plant operates at a reduced full power T-avg, the T' of the OTDT reactor trip function and the T" of the OPDT reactor trip function are set to be consistent with the reduced T-avg value. This effectively generates significant DNB margin and/or operating margin because the DNB limits are independent of the operating full power T-avg. With the OTDT and OPDT setpoint parameter values in the COLR, a licensee could take advantage of this margin in several different ways.

One example would be to develop a relationship between the full power T-avg and the OTDT setpoint K1 gain. This could be accomplished using the existing NRC approved methodology for calculating the OTDT and OPDT setpoints presented in Reference 6. Another relationship which could be determined would be between the OTDT/OPDT and core peaking factors. With reduced peaking factors, the OTDT/OPDT trip setpoints could be relaxed which would yield benefits through margin recovery.

The important consideration is that margin that would otherwise be unnecessarily allocated and "lost" when operating at a reduced T-avg, minimal steam generator tube plugging levels, or reduced peaking factors, could be utilized to enhance plant operating margins, enhance the OTDT and/or OPDT setpoints, and/or increase the flexibility of the core designs without any reduction in the margin to safety.

4.0 Replacement of the Reactor Core Safety Limits Figure

4.1 Basis for the Reactor Core Safety Limits Figure

The Technical Specification Reactor Core Safety Limits figure presents the limiting RCS temperature conditions (T-avg) as a function of pressurizer pressure and fractional rated thermal power. The figure was included in the Standard TS to satisfy the requirements of 10CFR50.36 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 RCSL figure is intended to provide the relationship between the process variables that are available to the operators, i.e., RCS temperature (T-avg), pressurizer pressure and rated thermal power level as defined by the excore detectors, and the DNB design basis. If a Condition I or II event were to occur, the safety limits figure could supposedly be used by the licensee to determine whether or not the DNB design basis was met. For Condition III and IV events, the RCSL figure is not applicable.

In addition to the above, the RCSLs figure is used in the generation of the OTDT and OPDT reactor trip setpoints. This is described in detail in Reference 6. It is for this reason that the figure contains hot-leg boiling limits, which are not true safety limits. The hot-leg boiling limits preclude saturation conditions and, thus help ensure that the measured ΔT remains proportional to the thermal power. The DNB limits of the figure are based on the DNBR safety analysis limit and assume a specific RCS total flow rate and a symmetrical reference axial power shape. Based on this figure, the gains (K1 through K6) of the OTDT and OPDT reactor trip setpoints are generated. For non-symmetrical power shapes that are more limiting than the reference axial power shape, the $f(\Delta I)$ penalty function of the OTDT reactor trip setpoint and the $f(\Delta I)$ penalty function of the OPDT reactor trip setpoint, if applicable, reduce the corresponding trip setpoints. Thus, the OTDT and OPDT reactor trip setpoints not only ensure that the RCSL figure is satisfied during a Condition I or II type event, they also ensure that for non-symmetrical axial power shapes (for which the RCSL figure is not necessarily applicable) that the DNB design basis is satisfied. Since the OTDT and OPDT setpoints are based on the RCSL figure, the only way to violate the figure is under the postulated condition where the reactor protection system does not function as designed. This is an important point since the operation of the reactor protection system and main steam system safety valves, will ensure that the DNB design basis is satisfied (as well as other applicable safety criteria) for any Condition I or II transient, independent of the RCSL figure.

4.2 Basis for Replacing the Reactor Core Safety Limits Figure

The first and most significant reason for replacing the RCSLs figure is that there are limitations associated with the figure that could result in a licensee drawing an incorrect conclusion with respect to the DNB design basis. Drawing a correct conclusion with respect to the safety limits is an important consideration given the ramifications associated with the violation of a safety limit. As noted above, the violation of a safety limit could only result if the reactor protection system were not functioning as designed. In this rare situation, using the RCSL figure to determine whether or not the DNB design basis had been violated or not has the following limitations.

A. As noted previously, the RCSL figure forms the basis for the OTDT and OPDT setpoints. Since these trip functions rely on the measured ΔT as an indication of power, it is important that hot-leg boiling is precluded to ensure that ΔT remains proportional to the reactor power. The hot-leg boiling limits are the less sloped lines towards the left side of the figure and are not true safety limits. Thus, a violation of the hot-leg boiling limits of the RCSL figure does not necessarily mean a safety limit has been violated.

B. The RCSL figure assumes all reactor coolant pumps are operating. If a partial or complete loss of flow transient occurs, the figure is not valid for this condition. Thus, comparing the most limiting temperature, pressure, and power condition under a loss of flow event to the RCSL figure could lead to an incorrect conclusion with respect to the DNB design basis since the figure is not valid under reduced RCS flow conditions.

C. The DNBR limit lines presented in the figure are based on a reference axial power shape. If the axial power shape during a transient were to become more limiting than the assumed reference axial power shape, comparing the most limiting transient temperature, pressure, and power condition to the RCSL figure could lead to an invalid conclusion as to whether or not the DNB design basis is satisfied. For instance, in the event of an Uncontrolled RCCA (Bank) Withdrawal at Power accident, the axial power shapes tend to become more positive (top skewed) as the rods are withdrawn from the core. If the resulting power shape is more limiting than that used to generate the RCSLs, the figure would not be valid. Conversely, if the transient axial power shape was not as limiting as that used to define the DNBR limits of the figure, the figure would not truly represent the limiting conditions with respect to the DNB safety limit. For these reasons, comparing the limiting temperature, pressure and power condition to the RCSL figure would not necessarily lead to the correct conclusion with respect to the DNB safety limit.

D. The RCSLs typically have DNBR margin built into them. Thus, a violation of the safety limits does not necessarily indicate that the plant has violated the licensed DNB design basis which is the true "safety limit".

In addition to the above, there are a number of other reasons why using the RCSLs may result in an incorrect conclusion with respect to the licensed DNB design basis, some of which are event specific and some of which are reload specific. For instance, the actual $F\Delta H$ may be significantly less than that used to generate the RCSL figure, the indicated conditions were actually reading high with respect to the actual conditions, the actual RCS total flow rate is greater than the Technical Specification limit, etc.

In summary, the reactor protection system and main steam system safety valves ensure that all the safety limits will be met, independent of the RCSL figure. Using the figure to determine whether or not a safety limit had been violated is marginal at best. Therefore, it is concluded that the figure is not needed. In the event of a Condition I or II transient, verification that the reactor protection system and the main steam system safety valves are functioning as designed will ensure that all safety limits are met. In the very low probability occurrence that the reactor protection system is not functioning as designed, an evaluation of any transient condition would be required to determine whether or not the DNB design basis is satisfied.

In addition to the above, replacing the RCSLs figure would eliminate the possibility of the figure being misused to define an "acceptable" operating configuration. Using this figure could result in a plant being placed in an unanalyzed condition. Finally, since the figure is based on the FΔH, which is currently presented in the COLR for many plants, a change to the COLR FΔH could require a license amendment request to revise the figure thereby negating the benefit of having the FΔH in the COLR.

4.3 Benefits of Replacing the Reactor Core Safety Limits Figure

The primary benefit of replacing the RCSLs figure is that it would eliminate the possibility of reaching an incorrect conclusion concerning the very important question of whether or not a safety limit has been violated for a Condition I or II event. A secondary benefit is that the removal of the figure would prevent the possibility of misusing the figure to define an "acceptable" operating configuration. Finally, there is a third benefit which is the potential to eliminate the need for a license amendment request if the RCSLs needed to be revised, which could result, for example, from a change in the FΔH value or the RCS total flow rate.

4.4 Revised Safety Limits

It is proposed that the RCSL figure be replaced with the DNB design basis limit and the fuel centerline melting limit. Both of these limits are criteria that must be satisfied for all Condition I and II transients. As was noted in section 4.2, confirmation that the reactor protection system and of the main steam system safety valves 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.

It should be noted that using the DNB and FCM criteria, in combination with ensuring compliance with the TS prior to the initiation of an event, satisfies 10CFR50.36 and is also consistent with the safety limits presented in other vendor standard Technical Specifications. Appendix A contains markups of the proposed RCSLs TS.

5.0 Conclusions

This report provides justification to support the TS changes required to allow licensees to improve margin utilization by expanding current COLRs to include cycle-specific RCS related TS limits.
The relocation of selected RCS related TS limits to the COLR would allow licensees to present cycle-specific operating condition parameters in the COLR. Thus, the safety analyses could credit the actual cycle-specific operating configuration, which is what is currently done for the core reload designs. This approach is supported by the existing NRC-approved reload methodologies, such as the Westinghouse reload methodology described in Reference 5, which examines each of the DNB Parameter TS limits of RCS T-avg, RCS total flow rate and pressurizer pressure as well as the OTDT and OPDT setpoints and the supporting bases for the setpoints, on a cycle-specific basis. Relocating these TS limit parameter values to the COLR, consistent with what has been accomplished for the core related parameters currently in the COLR, would allow licensees the flexibility to utilize available margins to increase cycle operating margins and/or improve core reload designs. It would also bring consistency to the TS, the safety analyses, and how the plant control and protection systems are set for each cycle. In addition, resources would be saved by licensees by minimizing and/or eliminating "trivial" LAR submittals and the need for associated NRC reviews.

Finally, the proposed replacement of the RCSL figure with the DNB design basis and fuel centerline melting criteria would reduce the chance of licensees reaching an incorrect conclusion with respect to the safety limit criteria in the postulated situation where the reactor protection system and main steam safety valves were not functioning as designed following a Condition I or II event. It would also eliminate the possibility that the figure could be misused to define an "acceptable" operating configuration.

It is important to note that while this report presents markups for NUREG-1431 TS, the information and markups presented within are applicable in principle and can be applied, with appropriate modifications, to plants with "Custom" and NUREG-0452 format TS.

6.0 References

1. Letter from C. I. Grimes (NRC) to R. A. Newton (WOG) titled "Acceptance for Referencing of Topical Report WCAP-14040, Revision 1, 'Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves' (TAC # M91749)," dated October 16, 1995.

2. Docket Nos. 50-413 and 50-414, R. E. Martin (NRC) to D. L. Rehn (Duke Power) titled "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)," dated March 25, 1994.

   Docket Nos. 50-369 and 50-370, V. Nerses (NRC) to T. C. McMeekin (Duke Power) "Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2 (TAC Nos. M85474 and M85475)," dated May 31, 1994.

4. Docket Nos. 50-282 and 50-306, D. C. Dilanni (NRC) to T. M. Parker (NSPC) titled "Amendment Nos. 92 and 85 to Facility Operating License Nos. DPR-42 and DPR-60: Relocation of Cycle Specific Operating Parameters from the Technical Specification to Core Reporting Limits Reports (TAC Nos. 75314 and 75315)," dated March 13, 1990.

5. Davidson, S. L. (Ed.), et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A (Westinghouse Proprietary Class 2), July 1985 (or other applicable reload methodology).

6. Ellenberger, S. L., et. al., "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," WCAP-8745-P-A, September 1986 (or other applicable Overtemperature and Overpower setpoint calculational methodology).

7. Christensen, J. A., et. al., "Melting Point of Irradiated Uranium Dioxide," Transactions of the American Nuclear Society, 7, No. 2, 1964.

Appendix A

Sample Technical Specification Markups

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 SLs specified in Figure 2.1.1-1~~.

Insert 1

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.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the [Plant Superintendent and Vice President – Nuclear Operations].

2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President – Nuclear Operations].

2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.

Insert 1

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ [1.17 for the WRB-1/WRB-2 DNB correlations].

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < [5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup].

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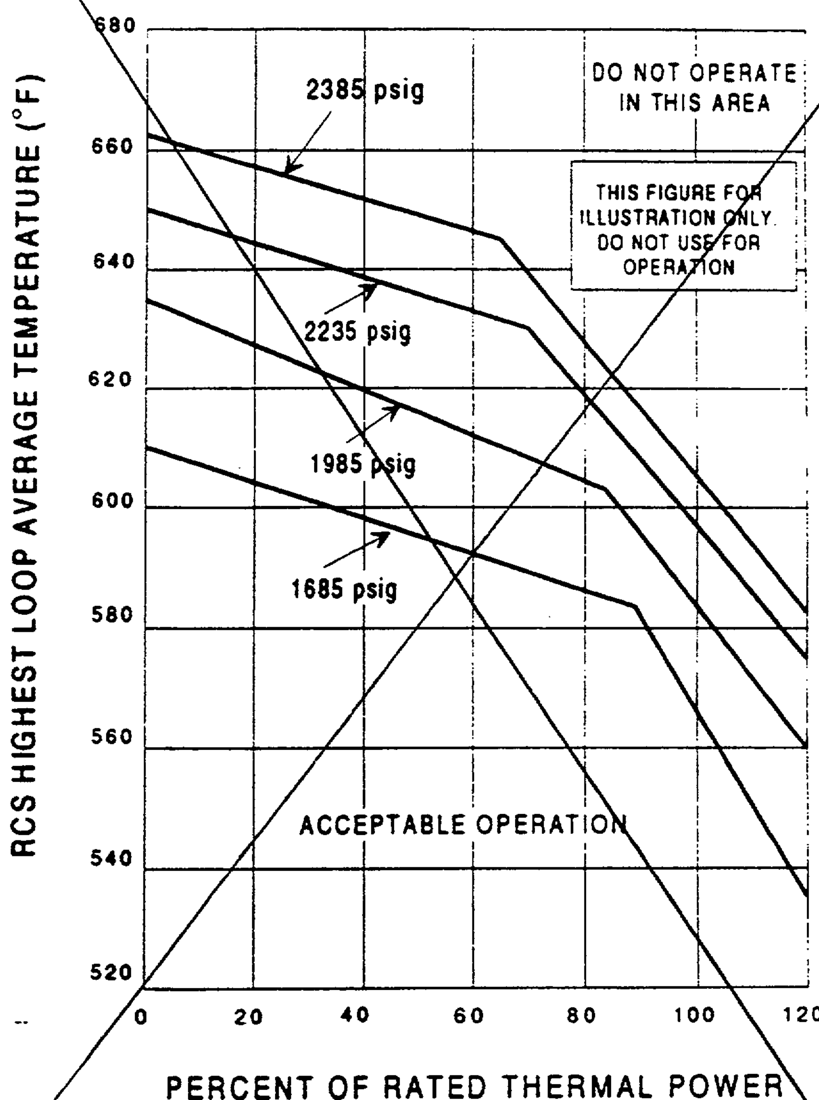

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

BASES

BACKGROUND (continued)

The proper functioning of the Reactor Protection System (RPS) 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 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 setpoints (Ref. 2), 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.

Automatic enforcement of these reactor core SLs is provided by the ~~following functions:~~ appropriate operation of the RPS and the steam generator safety valves.

~~a. High pressurizer pressure trip;~~

~~b. Low pressurizer pressure trip;~~

~~c. Overtemperature ΔT trip;~~

~~d. Overpower ΔT trip;~~

~~e. Power Range Neutron Flux trip; and~~

~~f. Steam generator safety valves.~~

~~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.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or 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

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

~~The curves are based on enthalpy hot channel factor limits provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.~~

~~The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).~~

Insert 2

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 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. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4,

(continued)

Insert 2

The reactor core SLs are established to preclude 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 RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature 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 RPS 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.

BASES

APPLICABILITY (continued)

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.

2.2.1

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.

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. ~~2~~ 3).

2.2.4

If SL 2.1.1 is violated, the Plant Superintendent and the Vice President – Nuclear Operations shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.5

If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. ~~3~~ 4). A copy of the report shall also be provided to the Plant Superintendent and the Vice President – Nuclear Operations.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.6

If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. FSAR, Section [7.2].

~~3. WCAP-8746-A, March 1977.~~

~~4. WCAP-9273-NP-A, July 1985.~~

3 ~~5~~. 10 CFR 50.72.

4 ~~6~~. 10 CFR 50.73.

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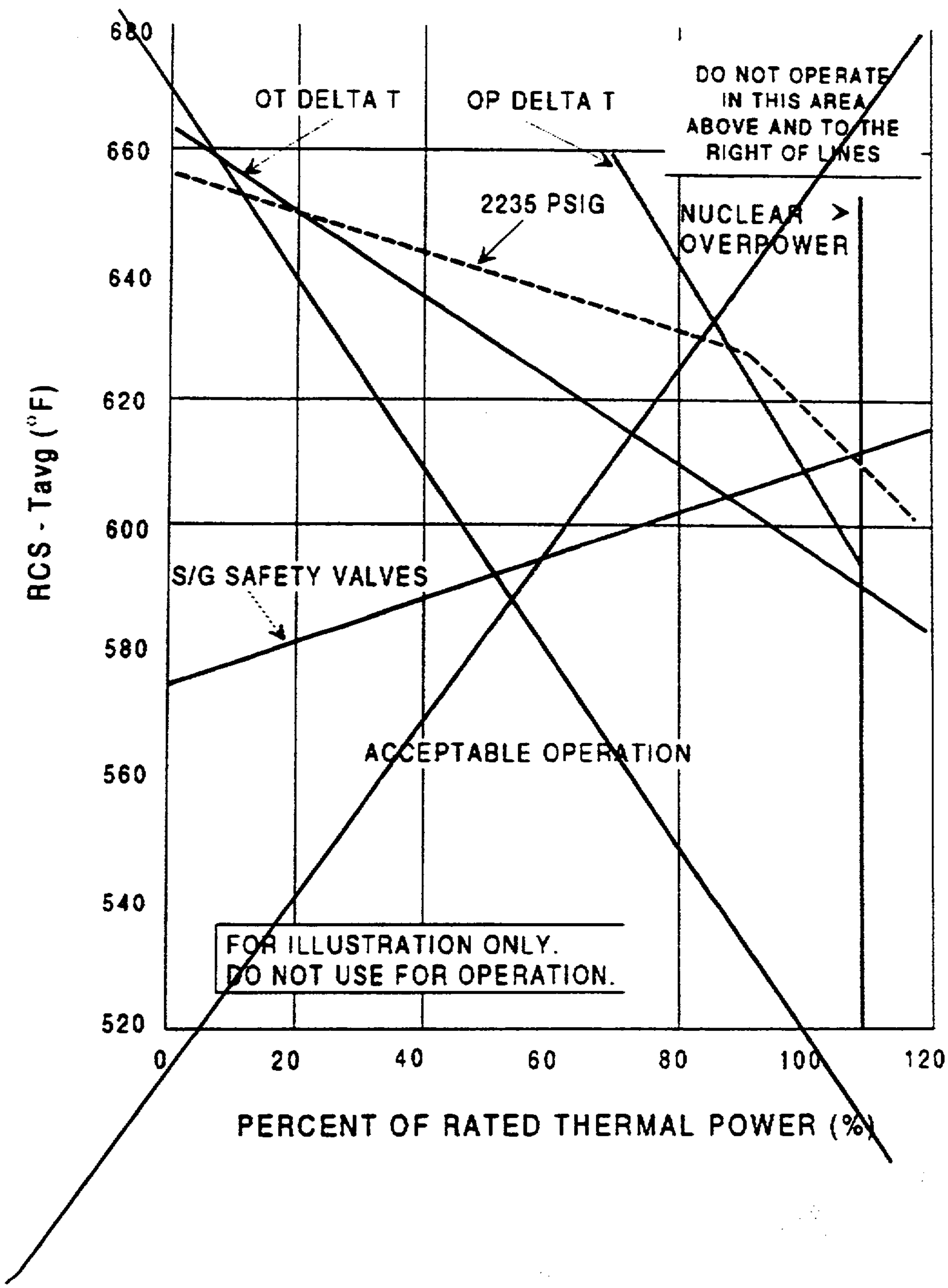

Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

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 Trip Setpoint by more than [3.8]% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)}\left[T \frac{1}{(1+\tau_6 s)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

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, ≤ [~~588~~]°F.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, ≤ [~~2235~~] psig

K_1 ≤ [~~1.09~~] K_2 ≥ [~~0.0138~~]/°F K_3 = [~~0.000671~~]/psig
τ_1 ≥ [~~8~~] sec τ_2 ≤ [~~3~~] sec τ_3 ≤ [~~2~~] sec
τ_4 ≥ [~~33~~] sec τ_5 ≤ [~~4~~] sec τ_6 ≤ [~~2~~] sec

$f_1(\Delta I)$ = ~~1.26~~(~~35~~ + $(q_t - q_b)$) when $q_t - q_b$ ≤ ✗ [~~35~~]% RTP
0% of RTP when -[~~35~~]% RTP < $q_t - q_b$ ≤ [~~7~~]% RTP
-~~1.05~~{$(q_t - q_b)$ - ~~7~~} when $q_t - q_b$ > [~~7~~]% 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.

– Deleted values will be presented in the COLR and replaced with an "*".

* as 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 Trip Setpoint by more than [3]% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{1+\tau_6 s} - T'' \right] - f_2(\Delta I) \right\}$$

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, ≤ [~~508~~]°F.

$K_4 \le$ [~~1.09~~] $K_5 \ge$ [~~0.02~~]/°F for increasing T_{avg}, [~~0~~]/°F for decreasing T_{avg} $K_6 \ge$ [0~~.00128~~]/°F when T > T", [~~0~~]/°F when T ≤ T"

$\tau_1 \ge$ [~~0~~] sec $\tau_2 \le$ [~~3~~] sec $\tau_3 \le$ [~~2~~] sec

$\tau_6 \le$ [~~2~~] sec $\tau_7 \ge$ [~~10~~] sec

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

- Deleted values will be presented in the COLR and replaced with an "*"

* as 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:~~ in the COLR.

~~a. Pressurizer pressure ≥ [2200] psig;~~

~~b. RCS average temperature ≤ [581]°F; and~~

~~c. RCS total flow rate ≥ [284,000] 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 within ~~≥ [2200] psig.~~ the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is within ~~≤ [581]°F.~~ the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within ~~≥ [284,000] gpm.~~ the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 24 hours after ≥ [90]% RTP. -- Verify by precision heat balance that RCS total flow rate is ~~≥ [284,000] gpm.~~ within the limit specified in the COLR.	[18] months

BASES

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion ~~of ≥ [1.3]~~. 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 is within the limits of LCO 3.1.7, "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 [2200] psig~~ and ~~the~~ RCS average temperature limit ~~of [581]°F~~ specified in the COLR correspond to the analytical limits ~~of [2205] psig and [595]°F~~ used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

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. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error ~~of [2.0]%~~ based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty ~~of [0.1]%~~ for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance ~~to [2.1]%~~ for no fouling.

Any fouling that might bias the flow rate measurement greater than ~~[0.1]%~~ the penalty for undetected fouling of the feedwater venturi 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.

(continued)

BASES

LCO (continued)

The ~~LCO~~ numerical values for pressure, temperature, and flow rate are given for the measurement location ~~but~~ have ~~not~~ been adjusted for instrument error. and specified in the COLR

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 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.

The DNBR limit ~~Another set of limits on DNB related parameters~~ is provided in SL 2.1.1, "Reactor Core SLs." ~~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. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

The conditions which define the DNBR limit

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 eliminate the potential for violation of the accident analysis bounds.

(continued)

COLR Analytical Methods for Inclusion in Specification 5.6.5

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (Westinghouse Proprietary Class 2) (or other applicable reload methodology).

2. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary Class 2) (or other applicable setpoint methodology).

Appendix B

Sample COLR Revisions

Appendix C

NRC RAI and WOG Response to RAI

Reactor Trip System Instrumentation Setpoints (Specification 3.3.1-1)

Overtemperature ΔT Setpoint Parameter Values

Parameter	Value
Overtemperature ΔT reactor trip setpoint	$K1 \leq 1.09$
Overtemperature ΔT reactor trip setpoint Tavg coefficient	$K2^* \geq 0.0138/°F$
Overtemperature ΔT reactor trip setpoint pressure coefficient	$K3^* \geq 0.000671/psi$
Indicated full power Tavg	$T' \leq 588°F$
Indicated pressurizer pressure	$P' \geq 2235$ psig
Measured reactor vessel ΔT lead/lag time constants	$\tau_1 \geq 8$ sec $\tau_2 \leq 3$ sec
Measured reactor vessel ΔT lag time constant	$\tau_3 \leq 2$ sec
Measured reactor vessel average temperature lead/lag time constants	$\tau_4 \geq 33$ sec $\tau_5 \leq 4$ sec
Measured reactor vessel average temperature lag time constant	$\tau_6 \leq 2$ sec
$f_1(\Delta I)$ "positive" breakpoint	+7% ΔI
$f_1(\Delta I)$ "negative" breakpoint	-35% ΔI
$f_1(\Delta I)$ "positive" slope	+1.05%/% ΔI
$f_1(\Delta I)$ "negative" slope	-1.26%/% ΔI

* These values should be set as close as reasonably possible to the nominal values to be consistent with the plant safety analyses.

Overpower ΔT Setpoint Parameter Values

Parameter	Value
Overpower ΔT reactor trip setpoint	$K4 \leq 1.09$
Overpower ΔT reactor trip setpoint Tavg rate/lag coefficient	$K5 \geq 0.02/°F$ for increasing Tavg $= 0/°F$ for decreasing Tavg
Overpower ΔT reactor trip setpoint Tavg heatup coefficient	$K6 \geq 0.00128/°F$ for $T > T''$ $= 0/°F$ for $T < T''$
Indicated full power Tavg	$T'' \leq 588°F$
Measured reactor vessel ΔT lead/lag time constants	$\tau_1 \geq 8$ sec $\tau_2 \leq 3$ sec
Measured reactor vessel ΔT lag time constant	$\tau_3 \leq 2$ sec
Measured reactor vessel average temperature lag time constant	$\tau_6 \leq 2$ sec
Measured reactor vessel average temperature rate/lag time constant	$\tau_7 \geq 10$ sec
$f_2(\Delta I)$ "positive" breakpoint	NA
$f_2(\Delta I)$ "negative" breakpoint	NA
$f_2(\Delta I)$ "positive" slope	NA
$f_2(\Delta I)$ "negative" slope	NA

DNB Parameters - Reactor Coolant System Tavg (Specification 3.4.1)

Parameter	Indicated Value
RCS average temperature	Tavg $\leq$ 581°F
Pressurizer pressure	Pressure $\geq$ 2200 psig
RCS total flow rate	Flow $\geq$ 284,000 gpm

Domestic Utilities
American Electric Power
Carolina Power & Light
Commonwealth Edison
Consolidated Edison
Duquesne Light
Duke Power
Georgia Power
Florida Power & Light
Houston Lighting & Power
New York Power Authority
Northeast Utilities
Northern States Power
Pacific Gas & Electric
Public Service Electric & Gas
Rochester Gas & Electric
South Carolina Electric & Gas
Southern Nuclear
Tennessee Valley Authority
TU Electric
Union Electric
Virginia Power
Wisconsin Electric Power
Wisconsin Public Service
Wolf Creek Nuclear

International Utilities
Electrabel
Kansai Electric Power
Korea Electric Power
Nuclear Electric plc
Nuklearna Elektrana
Spanish Utilities
Taiwan Power
Vattenfall

OG-98-118
November 25, 1998

WCAP-14483
Project Number 694

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

Attn: Chief, Information Management Branch
Division of Inspection and Support Programs

Subject: Westinghouse Owners Group
Response to NRC Request for Additional Information on WCAP-14483, "Generic Methodology for Expanded Core Operating Limits Report," (Non-Proprietary)
(MUHP1009)

Reference: 1) Westinghouse Owners Group Letter, T.V. Greene to Document Control Desk, "Transmittal of WCAP-14483 (Non-Proprietary), Generic Methodology for Expanded Core Operating Limits Report'," December 1, 1995.

2) NRC Letter, P. C. Wen to A. Drake, "Request for Additional Information for Westinghouse Topical Report WCAP-14483, 'Generic Methodology for Expanded Core Operating Limits Report'," September 2, 1998.

In December 1995 the Westinghouse Owners Group (WOG) submitted Westinghouse topical report WCAP-14483 (Non-Proprietary), "Methodology for Expanded Core Operating Limits Report," for NRC review (Reference 1). The NRC Staff has initiated their review of the topical report and issued a Request for Additional Information (RAI) (Reference 2). Attachment 1 provides the WOG response to the RAI.

Invoices associated with the review of this RAI response should be addressed to:

Mr. Andrew P. Drake, Project Manager
Westinghouse Owners Group
Westinghouse Electric Company
(Mail Stop 5-16E)
P.O. Box 355
Pittsburgh, PA 15230-0355

If you require further information, feel free to contact Mr. Ken Vavrek in the Westinghouse Owners Group Project Office at 412-374-4302.

Very truly yours,

Louis F. Liberatori Jr., Chairman
Westinghouse Owners Group

attachments/enclosures

98og118.doc

OG-98-118
November 25, 1998

cc: Steering Committee (1L, 1A)
Primary Representatives (1L, 1A)
Analysis Subcommittee Representatives (1L, 1A)
Licensing Subcommittee Representatives (1L, 1A)
P. Wen, USNRC (1L, 1A)
L. Kopp, USNRC OWFN 8 E23 (1L, 1A)
H. A. Sepp, W - ECE 4-07a(1L, 1A)
A. P. Drake, W - ECE 5-16 (1L, 1A)

Attachment 1

Response To NRC Request For Additional Information on WCAP-14483, "Generic Methodology For Expanded Core Operating Limits Report"

RAI Question 1

It is recognized that DNB and fuel centerline melt are the true safety limits. However, since these are not measurable quantities, they do not meet the requirements of 10CFR 50.36 which states that technical specification safety limits are limits upon important process variables. Justify how the requirements of 10 CFR 50.36 would still be met if Figure 2.1.1-1 were deleted.

Response:

Technical Specification Figure 2.1.1-1 provides a relationship between the process variables of Tavg, pressurizer pressure, and rated thermal power, and the DNB design basis limit. This Figure is a representation of "reactor core limits" and is not a complete representation of reactor core safety limits. The Figure can change on a cycle specific basis due to changes to $F\Delta H$ and RCS flow rate. As discussed in WCAP-14483, and stated above, the reactor core safety limits are DNB design basis limit and fuel centerline melt. The Reactor Protection System and the reactor core limits (Figure 2.1.1-1) would be used to determine whether the actual reactor core safety limits (DNB design basis limit and fuel centerline melt) were violated, should an event occur that could potentially challenge them. The Applicable Safety Analyses Section of the Bases of Technical Specification 2.1.1, contained in WCAP-14483, discusses verification of reactor core Safety Limits. This Figure is not used by the operators during plant operation. The relevant DNB process variables used by the operators are addressed by the Technical Specification requirements for RCS pressure, temperature, and flow.

To ensure that the requirements of 10CFR50.36 are met, i.e., limits upon important process variables, it is proposed that the <u>requirement</u> for a Reactor Core Limits Figure (2.1.1-1) be retained in the Technical Specifications, but contained in the Core Operating Limits Report (COLR), consistent with Generic Letter 88-16. The methodology used to calculate the reactor core limits figure is contained in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July 1985.

RAI Question 2

Some plants operate with lower steam generator tube plugging levels and thus higher RCS flow rates compared to what is assumed in the safety analyses. However, 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. We therefore recommend that if RCS flow rate is relocated to the COLR, the minimum limit for RCS total flow based on a staff approved analysis (e.g., maximum tube plugging) should be retained in the Technical Specifications similar to what is done for the positive limit on moderator temperature coefficient.

Response:

The WOG agrees to retain a minimum limit for RCS total flow rate in the Technical Specifications. This is consistent with the intent of WCAP-14483 to relocate the cycle specific RCS total flow rate to the COLR.

Westinghouse Electric Corporation **Energy Systems**

Box 355
Pittsburgh Pennsylvania 15230-0355

WOG-98-182

September 10, 1998

To: Westinghouse Owners Group Primary Representatives (1L, 1A)
Westinghouse Owners Group Analysis Subcommittee Representatives (1L, 1A)
Westinghouse Owners Group Licensing Subcommittee Representatives (1L, 1A)

Subject: Westinghouse Owners Group
<u>NRC Questions on WCAP-14483, "Generic Methodology for Expanded Core Operating Limits Report" (MUHP1009)</u>

Attached is a copy of the NRC letter, "Request for Additional Information for Westinghouse Topical Report WCAP-14483, Generic Methodology for Expanded Core Operating Limits Report", dated September 2, 1998. The WOG Analysis and Licensing Subcommittees will be preparing responses to the NRC questions.

If you have any questions regarding the NRC request or the WOG program please call Ken Vavrek, WOG Project Engineer, at 412-374-4302.

Very truly yours

Andrew P. Drake, Project Manager
Westinghouse Owners Group

attachment

cc: Steering Committee (1L, 1A)
K.J. Vavrek - ECE 5-16 (1L, 1A)
D. Huegel - ECE 427G (1L, 1A)
D. Hill - ECE 465A (1L, 1A)
J. Andrachek - ECE 413H (1L, 1A)

98wog182.doc

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 2, 1998

Mr. Andrew Drake
Westinghouse Owners Group
Westinghouse Electric Corporation
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, PA 15320-0355

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR WESTINGHOUSE TOPICAL REPORT WCAP-14483, "GENERIC METHODOLOGY FOR EXPANDED CORE OPERATING LIMITS REPORT"

Dear Mr. Drake :

By letter dated December 1, 1995, the Westinghouse Owners Group (WOG) submitted Westinghouse topical report WCAP-14483 for NRC review. In a recent telephone conversation, the WOG requested the staff to complete the review expeditiously. The staff has now completed the preliminary review and has determined a need for additional information. The attachment to this letter identifies the information required. Please address your response to the NRC Document Control Desk and reference WOG Project No. 694.

If you have any questions, please contact me at 301/415-2832 (email, pxw@nrc.gov) or Larry Kopp at 301/415-2879 (email, lik@nrc.gov).

Sincerely,

Peter C. Wen

Peter C. Wen, Project Manager
Generic Issues and Environmental
Projects Branch
Office of Nuclear Reactor Regulation

Enclosure: Questions on Topical
WCAP-14483

RECEIVED
SEP 08 1998
WOG PROJECT OFFICE

REQUEST FOR ADDITIONAL INFORMATION

TOPICAL REPORT WCAP-14483

1) It is recognized that DNB and fuel centerline melt are the true safety limits. However, since these are not measurable quantities, they do not meet the requirements of 10 CFR 50.36 which states that technical specification safety limits are limits upon important process variables. Justify how the requirements of 10 CFR 50.36 would still be met if Figure 2.1.1-1 were deleted.

2) Some plants operate with lower steam generator tube plugging levels and thus higher RCS flow rates compared to what is assumed in the safety analyses. However, 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. We therefore recommend that if RCS flow rate is relocated to the COLR, the minimum limit for RCS total flow based on a staff approved analysis (e.g., maximum tube plugging) should be retained in the technical specifications similar to what is done for the positive limit on moderator temperature coefficient.

Enclosure