WOLF CREEK
NUCLEAR OPERATING CORPORATION

Richard A. Muench
Vice President Engineering
and Information Services

APR 3 2001

ET 01-0008

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station: P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Relocation of Technical Specification Cycle Specific Parameters to the Core Operating Limits Report

Gentlemen:

Wolf Creek Nuclear Operating Corporation (WCNOC) herewith transmits an application for amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS).

The proposed changes relocate Reactor Coolant System (RCS) related cycle-specific parameter limits from the Technical Specifications to, and thus expanding, the Core Operating Limits Report (COLR) for Wolf Creek Generating Station. The justification to implement the expansion of the COLR is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report." Additionally, TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is revised to allow topical reports to be identified by title and number only. The proposed changes will allow WCNOC the flexibility to enhance plant operating margin and core design margin, consistent with 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 TS 5.6.5, COLR."

The proposed changes to TS 2.1.1, "Reactor Core Safety Limits," and associated Bases relocate the reactor core safety limit figure to the COLR and replace it with the more specific fuel DNB Ratio (DNBR) and peak fuel centerline temperature safety limit requirements. The proposed changes to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," relocate the Overtemperature ΔT and Overpower ΔT constant (K) values, the dynamic compensation time (τ) values, and the breakpoint and slope values for the $f(\Delta I)$ penalty functions to the COLR. Additionally, the proposed changes to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and associated Bases relocate the pressurizer pressure, RCS average temperature, and RCS total flow rate values to the COLR. However, as discussed in WCAP-14483-A, the minimum limit for RCS total flow rate is being retained in TS 3.4.1.

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NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 4, 1988, provides guidance to licensees for the removal of such cycle-dependent variables from the 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. The changes WCNOC proposes herein meet these criteria. The proposed changes will obviate the need for future revisions of the TS to change the value of those operating limits which cannot be specified to reasonably bound several operating cycles without incurring a significant loss of operating flexibility.

The WCNOC Plant Safety Review Committee and the Nuclear Safety Review Committee have reviewed this amendment application. Attachments I through VI provide the required affidavit, description of proposed license changes and assessment, existing marked-up TS page, revised TS page, proposed TS Bases changes (provided for information only), summary of regulatory commitments made in this submittal.

WCNOC requests approval of the proposed license amendment by October 31, 2001, with the amendment being implemented prior to startup from Refueling Outage 12. Implementation prior to startup from Refueling Outage 12 would not require the issuance of a mid-cycle revision to the COLR. An expanded revised COLR will be issued concurrent with implementation of the approved TS amendment requested herein.

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 assessment need be prepared in connection with the issuance of this amendment.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State Official. If you should have any questions regarding this submittal, please contact me at (620) 364-4034, or Mr. Tony Harris at (620) 364-4038.

Very truly yours,

Richard A. Muench

RAM/rlr

Attachments: I - Affidavit
II - Description and Assessment
III - Markup of Technical Specification pages
IV - Retyped Technical Specification pages
V - Proposed Bases Changes (for information only)
VI - List of Commitments

cc: V. L. Cooper (KDHE), w/a
J. N. Donohew (NRC), w/a
W. D. Johnson (NRC), w/a
E. W. Merschoff (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Richard A. Muench, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering and Information Services of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By [signature]
Richard A. Muench
Vice President Engineering
and Information Services

SUBSCRIBED and sworn to before me this 3rd day of April , 2001.

[signature]
Notary Public

NOTARY PUBLIC STATE OF KANSAS CINDY NOVINGER My Appt. Exp. 7/8/02

Expiration Date July 8, 2002

ATTACHMENT II

DESCRIPTION AND ASSESSMENT

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

1.1 The proposed changes relocate Reactor Coolant System (RCS) related cycle-specific parameter limits from the Technical Specifications (TS) to, and thus expanding, the Core Operating Limits Report (COLR) for Wolf Creek Generating Station. The justification to implement the expansion of the COLR is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," (Reference 1). Additionally, TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is revised to allow topical reports to be identified by title and number only. The proposed changes will allow WCNOC the flexibility to enhance plant operating margin and core design margin 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" (Reference 9) and TSTF-363, Revision 0, "Relocate Topical Report References in TS 5.6.5, COLR" (Reference 10).

1.2 Updated Safety Analysis Report (USAR) Section

No changes to the USAR are currently anticipated as a result of this license amendment request.

2.0 DESCRIPTION

<u>Reactor Core Safety Limits</u>

TS Figure 2.1.1-1, "Reactor Core Safety Limits," is being relocated to the COLR, and is being replaced with more specific fuel departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature safety limits.

As documented in the Westinghouse Owners Group (WOG) response (Reference 8) to the NRC Request for Additional Information (Reference 7) associated with Reference 1, because the Reactor Core Safety Limit figure is being relocated to the COLR, the "requirement" for this Reactor Core Safety Limit figure will be retained in the TS. The methodology used to calculate the Reactor Core Safety Limit figure is contained in WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station," (Reference 4) as approved by the NRC on March 10, 1993 (Reference 5). The existing TS Section 5.6.5b, COLR Reporting Requirements, specifically requires Reference 4 to be used as one of the analytical methods to determine the core operating limits. Therefore, the NRC's request (Reference 7) that the NRC-approved methodology used to derive the parameters in the figure should be referenced in the Reporting Requirements section of the TS, is currently in-place.

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

TS Table 3.3.1-1, Note 1: OTΔT, and Note 2: OPΔT,setpoint parameter constant (K_1 through K_6) values, dynamic compensation time (τ_1 through τ_7) values, T′, T″, P′, and the breakpoint and slope values for the f(ΔI) penalty functions for the trip setpoints are being relocated to the COLR.

RCS Pressure, Temperature and Flow DNB Parameters

The TS 3.4.1 limits specified for the RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate are being relocated to the COLR.

Because the RCS total flow rate is being relocated to the COLR, the minimum limit for RCS total flow rate of $\geq$ 371,000 gpm, based on NRC approved analysis, is being retained in TS 3.4.1. As documented in the NRC Request for Additional Information (RAI), (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 NRC. To comply with this recommendation and the Westinghouse Owners Group (WOG) response (Reference 8), the minimum limit for RCS total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than that reviewed by the NRC will not be used.

Additionally, the Note associated with the Applicability is modified for format correctness.

COLR Analytical Methods

An additional reference is being added to the listing of NRC reviewed and approved analytical methods used to determine the core operating limits provided in TS Section 5.6.5b, "CORE OPERATING LIMITS REPORT (COLR)." Specifically, Westinghouse WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," (Reference 6), is being added as this methodology is used by WCNOC in calculating setpoint uncertainties.

TS 5.6.5b. identifies the approved topical reports and analytical methods used to determine the core operating limits. This section is revised to specify the approved topical reports by number and title only consistent with TSTF-363, Revision 0, "Revise Topical Report references in ITS 5.6.5, COLR."

3.0 BACKGROUND

NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications" (Reference 3), provides guidance to licensees for the removal of such cycle-dependent variables from the TS provided that the values of these variables are included in a COLR and are determined in accordance with NRC-approved methodology which is referenced in the TS. The changes we are proposing herein meet these criteria. The proposed changes will obviate the need for frequent future revisions of the TS to change the values of those operating limits which cannot be specified to reasonably bound several operating cycles without incurring a significant loss of operating flexibility.

WCAP-14483, "Generic Methodology for Expanded Core Operating Limits," was submitted to the staff on December 1, 1995. The topical report provided 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, provided 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, proposed revising Section 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," of NUREG-1431, Revision 1, to allow Topical Reports to be identified by number and title. TSTF-363, Revision 0, was approved by the NRC on April 13, 2000.

RCS Pressure, Temperature and Flow DNB Parameters

The TS limits on the DNB parameters assure that pressurizer pressure, RCS average temperature, and RCS total flow rate 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 Updated Safety Analysis Report (USAR) accident analyses. For Condition I and II events for which precluding DNB is the primary 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 initiation of the event. The DNB parameter limits are also based on the initial conditions assumed for Condition III and IV events for which precluding DNB is not a criterion. Given that the DNB parameter limits ensure that the DNB design basis and other safety criteria are satisfied, continuous plant operation at less than limiting conditions would result in margin to these safety criteria.

Overtemperature ΔT and Overpower ΔT Parameters

The OTΔT and OPΔT reactor trip functions ensure that during any Condition I or II transient, there is at least a 95% probability that the peak kW/ft fuel rods will not exceed the uranium dioxide, UO_2, melting temperature. 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. 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 DNB design basis is defined as the probability being at least 95% at a 95% confidence level that DNB will not occur on the limiting fuel rod(s). 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 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 function is correlated with the vessel ΔT, the RCS average temperature, and pressurizer pressure. A compensating term which is a function of ΔI is also factored into the OTΔT setpoint to account for the affected changes in the axial power shape. The setpoint is scaled to be consistent with the full power operating conditions.

The OPΔT reactor trip function, in conjunction with the OTΔT reactor trip function, ensures operation within the fuel temperature design basis. This is accomplished through the OPΔT reactor 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 vessel average temperature, is factored into the calculated OPΔT tripsetpoint. The setpoint is set to be consistent with the nominal full power operating conditions.

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 fractional RATED 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 RCS average temperature, pressurizer pressure and RATED 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.

Because 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 limit and assume a specific RCS 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(ΔI) penalty functions reduce the corresponding tripsetpoints. 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 ensure that for non-symmetrical axial power shapes the DNB design basis is satisfied. Because 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 Protection System (RPS) do not function as designed. Operation of the RPS and the Main Steam Safety Valves (MSSVs) ensure that the DNB design basis is satisfied for any Condition I or II transient, independent of the Reactor Core Safety Limits figure.

COLR Analytical Methods

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

4.0 TECHNICAL ANALYSIS

The justification to implement the expansion of the COLR is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," as approved in the NRC Safety Evaluation Report dated January 19, 1999, (Reference 2). The changes to the Standard Technical Specifications was approved in TSTF-339, Revision 2.

The justification for allowing Topical Reports to be referenced by title and number was approved in TSTF-363, Revision 0.

RCS Pressure, Temperature and Flow DNB Parameters

Relocating the DNB parameters limit values to the COLR allows the flexibility to utilize the available margins to increase cycle operating margins and improve core reload designs without the requirement of cycle-specific license amendments. The relocation of the 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 that the plant control and protection systems are set to for a given cycle.

The proposed changes to the Applicability Note provides for consistency with NUMARC 93-03, "Writer's Guide for the Restructured Technical Specifications," and NUREG-1431, Revision 1, "Standard Technical Specifications Westinghouse Plants."

Overtemperature ΔT and Overpower ΔT Parameters

Relocation of the OTΔT and OPΔT setpoint parameter values to the COLR minimizes the chance that a reload-related parameter change would necessitate a TS change. In addition, significant DNB and operating margin currently utilized in the setpoints that is unnecessarily allocated and thus unavailable could be utilized to enhance plant operating margins, enhance the OTΔT and OPΔT setpoints, and increase the flexibility of the core designs without any reduction in the margin of safety.

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.

It is proposed that the Reactor Core Safety Limits figure be replaced with the DNB design basis limit and the fuel centerline melting limit as these limits are criteria that must be satisfied for all Condition I and II transients. Confirming that the RPS 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.

COLR Analytical Methods

NRC Generic Letter 88-16 allows removal of cycle-dependent variables from the TS provided that the values of these variables are included in a COLR and are determined with NRC reviewed and approved methodology which is referenced in the TS. Safety limits, however, may not be placed in the COLR. Therefore, the "requirement" for the Reactor Core Safety Limit figure is being retained in the TS by the current reference to the NRC reviewed and approved methodology (Reference 4) used to calculate the Reactor Core Safety Limit figure in TS Section 5.6.5b. The applicable NRC reviewed and approved setpoint methodology (Reference 6) for the OTΔT and OPΔT setpoint parameter values being relocated to the COLR is being added as a referenced analytical method in TS Section 5.6.5b.

TS 5.6.5b. identifies the approved topical reports and analytical methods used to determine the core operating limits. This section is revised to specify the approved topical reports by number and title only consistent with TSTF-363, Revision 0. In a letter dated December 15, 1999 (Reference 11), 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 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 would provide specific information identifying the particular approved Topical Report used to determine the core limits for the particular cycle in the COLR.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

WCNOC has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

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

   Response: No

   The proposed changes are programmatic and administrative in nature which do not physically alter safety related systems, nor affect the way in which safety related systems perform their functions. More specific requirements regarding the safety limits (i.e., DNBR limit and peak fuel centerline temperature limit) are being imposed in TS 2.1.1, "Reactor Core Safety Limits," which replace the Reactor Core Safety Limits figure and are consistent with the values stated in the USAR. The proposed changes remove cycle-specific parameter limits from TS 3.4.1 and relocate them to the COLR which do not change plant design or affect system operating parameters. In addition, the minimum limit for RCS total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than reviewed by the NRC will not be used. The proposed changes do not, by themselves, alter any of the parameter limits. The removal of the cycle-specific parameter limits from the TS does not eliminate existing requirements to comply with the parameter limits. The existing TS Section 5.6.5b, COLR Reporting Requirements, continues to ensure that the analytical

methods used to determine the core operating limits meet NRC reviewed and approved methodologies. The existing TS Section 5.6.5c, COLR Reporting Requirements, continues 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 would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

Although the relocation of the cycle-specific parameter limits to the COLR would allow revision of the affected parameter limits without prior NRC approval, there is no significant effect on the probability or consequences of an accident previously evaluated. Future changes to the COLR parameter limits could result in event consequences which are either slightly less or slightly more severe than the consequences for the same event using the present parameter limits. The differences would not be significant and would be bounded by the existing requirement of TS Section 5.6.5c to meet the applicable limits of the safety analyses.

The cycle-specific parameter limits being transferred from the TS to the COLR will continue to be controlled under existing programs and procedures. The USAR accident analyses will continue to be examined with respect to changes in the cycle-dependent parameters obtained using NRC reviewed and approved reload design methodologies, ensuring that the transient evaluation of new reload designs are bounded by previously accepted analyses. This examination will continue to be performed pursuant to 10 CFR 50.59 requirements ensuring that future reload designs will not involve a significant increase in the probability or consequences of an accident previously evaluated. Additionally, the proposed changes do not allow for an increase in plant power levels, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not change the types or increase the amounts of any effluents released offsite.

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

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

Response: No

The proposed changes that retain the minimum limit for RCS total flow rate in the TS, and that relocate certain cycle-specific parameter limits from the TS to the COLR, thus removing the requirement for prior NRC approval of revisions to those parameters, do not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no changes being made to the parameters within which the plant is operated, other than their relocation to the COLR. There are no setpoints affected by the proposed changes at

which protective or mitigative actions are initiated. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

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 would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

Relocation of cycle-specific parameter limits has no influence or impact 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 methodology. The proposed changes do not alter assumptions made in the safety analysis and operation within the core operating limits will continue.

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

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

Response: No

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes do not physically alter safety-related systems, nor does it effect the way in which safety-related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed changes. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. As the proposed changes to relocate cycle-specific parameter limits to the COLR will not affect plant design or system operating parameters, there is no detrimental impact on any equipment design parameter, and the plant will continue to operate within prescribed limits.

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 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 would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

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

Based on the above evaluations, WCNOC concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

5.2 Regulatory Safety Analysis

Applicable Regulatory Requirements/Criteria

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 in the COLR. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (TS)," provided 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 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.

Conclusion

The regulatory requirements/criteria continue to be met. The proposed changes are in compliance with 10 CFR 50.36 and meet the guidance contained in Generic Letter 88-16.

6.0 ENVIRONMENTAL EVALUATION

WCNOC has determined that the proposed amendment would not change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, nor would it change an inspection or surveillance requirement. WCNOC has evaluated the proposed changes and has determined that the changes do not involve (i) a

significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

7.0 REFERENCES

1. Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," November 1995.

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,'" January 19, 1999.

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

4. WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."

5. "Wolf Creek Nuclear Operating Corporation - Reload Safety Evaluation Methodology for Wolf Creek Generating Station (TAC NO. M82970)," March 10, 1993.

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

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

8. L. F. Liberatori Jr. (WOG) Response to Request for Additional Information to NRC Document Control Desk, November 25, 1998.

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

10. TSTF-363, Revision 0, "Relocate Topical Report References in TS 5.6.5, COLR."

11. 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.

8.0 PRECEDENTS

There are precedents for relocating RCS related cycle-specific parameter limits for the Technical Specifications to the COLR. These precedents are based on the justification provided in WCAP-14483-A (Reference 1). The operating licenses for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, have been amended to relocate cycle-specific parameter limits to the COLR. These amendment, Nos. 106 and 113, respectively, were issued on May 15, 2000.

ATTACHMENT III

MARKUP OF TECHNICAL SPECIFICATION PAGES

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

limits

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~~.

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

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.23 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

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.

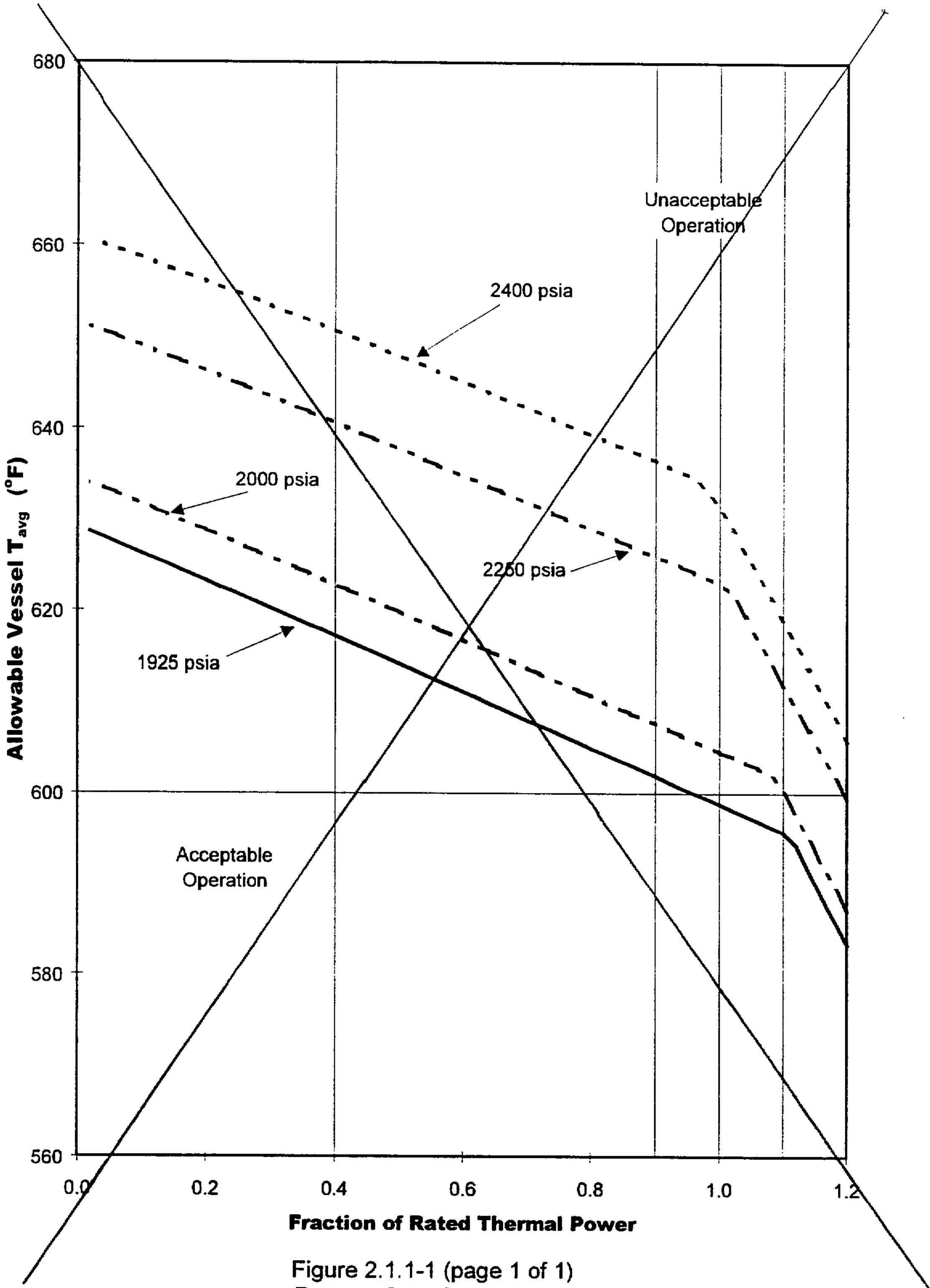

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.3% of ΔT span.

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

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, ≤ 586.5°F. *

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure ≥ 2235 psig.

K_1 = 1.15 *, τ_1 = 8 sec, τ_4 = 16 sec

K_2 = 0.0137/°F *, τ_2 = 3 sec, τ_5 = 4 sec

K_3 = 0.000671/psig *, τ_3 = 2 sec, τ_6 = 0 sec

$f_1(\Delta I)$ = *

- 0.0227 {23% + $(q_t - q_b)$} when $q_t - q_b$ < -23% RTP
- 0% of RTP when -23% RTP ≤ $q_t - q_b$ ≤ 5% RTP
- 0.0184 {$(q_t - q_b)$ - 5%} when $q_t - q_b$ > 5% 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 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 2.6% of ΔT span.

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

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 indicated T_{avg} at RTP (Calibration temperature for ΔT instrumentation), ≤ ~~586.5~~ °F. *

K_4 = ~~1.10~~ *

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

K_6 = ~~0.00128~~ /°F when T > T″ *
~~0~~ /°F when T ≤ T″ *

τ_1 = ~~5~~ sec *
τ_2 = ~~3~~ sec *
τ_3 = ~~2~~ sec *
τ_6 = ~~0~~ sec *
τ_7 = ~~10~~ sec *

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

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 ~~≥ 2220 psig~~ is greater than or equal to the limit specified in the COLR

b. RCS average temperature ~~≤ 590.5°F~~ is less than or equal to the limit specified in the COLR; and

c. RCS total flow rate ≥ 37.1×10^4 gpm and greater than or equal to the limit specified in the COLR.

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. ------------NOTE------------ Not applicable to RCS total flow rate. ------------------------------ One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 ------------NOTE------------ THERMAL POWER does not have to be reduced to comply with this Required Action. -------------------------------- Perform SR 3.4.1.3.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ~~≥ 2220 psig~~ greater than or equal to the limit specified in the COLR.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.1.2	Verify RCS average temperature is ~~≤ 590.5°F~~ less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 37.1 x 10^4 gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	-----NOTE----- Not required to be performed until 7 days after ≥ 95% RTP. -----	
	Verify by precision heat balance that RCS total flow rate is ≥ 37.1 x 10^4 gpm and greater than or equal to the limit specified in the COLR.	18 months

5.6 Reporting Requirements (continued)

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 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

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. Specification 3.1.3: Moderator Temperature Coefficient (MTC),
2. Specification 3.1.5: Shutdown Bank Insertion Limits,
3. Specification 3.1.6: Control Bank Insertion Limits,
4. Specification 3.2.3: Axial Flux Difference,
5. Specification 3.2.1: Heat Flux Hot Channel Factor, $F_Q(Z)$,
6. Specification 3.2.2: Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$),
7. Specification 3.9.1: Boron Concentration, ~~and~~
8. SHUTDOWN MARGIN for Specification 3.1.1 and 3.1.4, 3.1.5, 3.1.6, and 3.1.8.~~.~~, (3)
9. Specification 3.3.1: Overtemperature ΔT and Overpower ΔT Trip Setpoints, and
10. Specification 3.4.1: Reactor Coolant System pressure, temperature, and flow DNB limits.

(continued)

5.6 Reporting Requirements

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. WCNOC Topical Report TR 00-0025 W01, ~~NRC Safety Evaluation Report dated October 29, 1992, for the~~ "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station" ~~(ET-90-0140, ET 92-0103)~~.

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

3. WCNOC Topical Report NSAG-006 ~~NRC Safety Evaluation Report dated September 30, 1993, for the~~ "Transient Analysis Methodology for the Wolf Creek Generating Station" ~~(ET-91-0026, ET 92-0142, WM 93-0010, WM 93-0028)~~.

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

5. WCNOC Topical Report NSAG-007, ~~NRC Safety Evaluation Report dated March 10, 1993, for the~~ "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station" ~~(ET 92-0032, ET 93-0017)~~.

6. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7" ~~(NA 92-0073, NA 93-0013, NA 93-0054)~~.

7. WCAP-10266-P-A, ~~NRC Safety Evaluation Report dated November 13, 1986, for~~ "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code" ~~(WCAP-10266-P-A, Rev. 2)~~.

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

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

-A,"

9. ~~NRC Safety Evaluation Report dated June 23, 1986, "Acceptance for Referencing of Topical Report~~ WCAP 10965-~~P and WCAP 10966-NP~~ ANC: A Westinghouse Advanced Nodal Computer Code."

-P-A

10. ~~NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report~~ WCAP-12610, ~~"~~VANTAGE+ Fuel Assembly Reference Core Report ~~(TAC NO. 77258)~~" ~~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).~~

"

11. WCAP-8745-P-A, "Design Bases for the Thermal Power ΔT and Thermal Overtemperature ΔT Trip Functions."

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 settings 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, "Low Temperature Overpressure Protection System."

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

(continued)

ATTACHMENT IV

RETYPED TECHNICAL SPECIFICATION PAGES

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 departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.23 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

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.

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 $\geq 37.1 \times 10^4$ gpm and greater than or equal to the limit specified in the COLR.

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. -------------NOTE------------- Not applicable to RCS total flow rate. --------------------------------- One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------NOTE------------- Required Action B.2 must be completed whenever Condition B is entered. ---------------------------------- RCS flow rate not within limits.	B.1.1	Restore RCS flow rate to within limits.	2 hours
	<u>OR</u>		
	B.1.2.1	Reduce THERMAL POWER to < 50% RTP.	2 hours
		<u>AND</u>	
	B.1.2.2	Reduce Power Range Neutron Flux - High trip setpoints to ≤ 55% RTP.	6 hours
		<u>AND</u>	
	B.1.2.3	Reduce THERMAL POWER to < 5% RTP.	74 hours
	<u>AND</u>		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 -------------NOTE------------- THERMAL POWER does not have to be reduced to comply with this Required Action. ---------------------------------- Perform SR 3.4.1.3.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
C. Required Action and associated Completion Time of Condition A or B not met.	C.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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
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 $\geq 37.1 \times 10^4$ gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------------NOTE------------------------- Not required to be performed until 7 days after $\geq$ 95% RTP. -- Verify by precision heat balance that RCS total flow rate is $\geq 37.1 \times 10^4$ gpm and greater than or equal to the limit specified in the COLR.	18 months

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.3% of ΔT span.

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

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, ≤ * °F.

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

K_1 = *	K_2 = */°F	K_3 = */psig
τ_1 = * sec	τ_2 = * sec	τ_3 = * sec
τ_4 = * sec	τ_5 = * sec	τ_6 = * sec

$f_1(\Delta I)$	=	* {*% - (q_t - q_b)}	when q_t - q_b < *% RTP
		*% 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 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 2.6% of ΔT span.

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

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 indicated T_{avg} at RTP (Calibration temperature for ΔT instrumentation), $\leq$ * °F.

K_4 = *

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

K_6 = */°F when T > T″
*/°F when T $\leq$ T″

τ_1 = * sec τ_2 = * sec τ_3 = * sec

τ_6 = * sec τ_7 = * sec

$f_2(\Delta I)$ = *

The values denoted with * are specified in the COLR.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

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.

(continued)

5.6 Reporting Requirements (continued)

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 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

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. Specification 3.1.3: Moderator Temperature Coefficient (MTC),
   2. Specification 3.1.5: Shutdown Bank Insertion Limits,
   3. Specification 3.1.6: Control Bank Insertion Limits,
   4. Specification 3.2.3: Axial Flux Difference,
   5. Specification 3.2.1: Heat Flux Hot Channel Factor, $F_Q(Z)$,
   6. Specification 3.2.2: Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$),
   7. Specification 3.9.1: Boron Concentration,
   8. SHUTDOWN MARGIN for Specification 3.1.1 and 3.1.4, 3.1.5, 3.1.6, and 3.1.8,
   9. Specification 3.3.1: Overtemperature ΔT and Overpower ΔT Trip Setpoints, and

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

10. Specification 3.4.1: Reactor Coolant System pressure, temperature, and flow DNB limits.

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. WCNOC Topical Report TR 00-0025 W01, "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station."

2. WCAP-11397-P-A, "Revised Thermal Design Procedure."

3. WCNOC Topical Report NSAG-006, "Transient Analysis Methodology for the Wolf Creek Generating Station."

4. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification."

5. WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."

6. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7."

7. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."

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."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

10. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."

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

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 settings 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, "Low Temperature Overpressure Protection System."

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

(continued)

ATTACHMENT V

PROPOSED BASES CHANGES

BASES

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 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.

appropriate operation of the RPS and the steam generator safety valves.

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

~~a. High pressurizer pressure trip;~~

~~b. Low pressurizer pressure trip;~~

~~c. Overtemperature ΔT trip;~~

~~d. Overpower ΔT trip; and~~

~~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 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 USAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

BASES

SAFETY LIMITS

The ~~curves~~ figure provided in ~~Figure 2.1.1-1 show~~ the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature below which the calculated DNBR is not less than the design DNBR values 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.

Insert 1

~~The curves are based on enthalpy 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 $F_1(\Delta I)$ 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).~~

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. 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

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.

INSERT 1

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 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

REFERENCES

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

2. USAR, Chapter 15.

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

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

BASES

APPLICABLE SAFETY ANALYSIS (continued)

distribution limits are satisfied per LCO 3.1.4, "Rod Group Alignment Limits; " LCO 3.1.5, "Shutdown Bank 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)."

specified in the COLR

The pressurizer pressure limit ~~of 2220 psig~~ and ~~the~~ RCS average temperature limit ~~of 590.5°F~~ correspond to the analytical limits ~~of 2205 psig and 593.0°F~~ used in the safety analyses, with allowance for measurement uncertainty.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins completely offset any rod bow penalties. This is the margin between the correlation DNBR limit and the safety analysis limit DNBR. These limits are specified in the COLR. The applicable values of rod bow penalties are referenced in the USAR.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(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. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on [maximum analyzed steam generator tube plugging], is retained in the TS LCO.

The RCS total flow rate limit contains a measurement error ~~of 2.5%~~ based on performing a precision heat balance and using the result to normalize 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.

The effect of any fouling that might bias the flow rate measurement shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The LCO numerical values for pressure, temperature, and flow rate specified in the COLR have been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS total 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.

BASES

APPLICABILITY (continued)

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 changes > 5% RTP per minute or a THERMAL POWER step ramp > 10% RTP. The pressure transient 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." The conditions which define the DNBR limit ~~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). Condition A is modified by a Note stating that this condition does not apply to RCS total flow rate.

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.1.1

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 to restore DNB margin and reduce the potential for violation of the accident analysis limits.

Condition B is modified by a Note stating that Required Action B.2 must be completed whenever Condition B is entered. Thus, if power is not reduced because Required Action B.1.1 is completed within the 2 hour time period, Required Action B.2 nevertheless requires verifying RCS total flow rate within 12 hours in accordance with SR 3.4.1.3.

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Tony Harris, Manager Regulatory Affairs at Wolf Creek Generating Station, (620) 364-4038.

COMMITMENT	Due Date/Event
The amendment will be implemented prior to startup from Refueling Outage 12.	Prior to startup from Refueling Outage 12
An expanded revised COLR will be issued concurrent with implementation of the approved TS amendment requested herein.	Upon implementation of the license amendment